CHAPTER 59

PRA RESULTS AND INSIGHTS

59.1 Introduction

This chapter summarizes the use of the AP1000 PRA in the design process, PRA results and insights, plant features important to reducing risk, and PRA input to the design certification process.

AP1000 is expected to achieve a higher standard of severe accident safety performance than current operating plants, because both prevention and mitigation of severe accidents have been addressed during the design stage, taking advantage of PRA insights, PRA success criteria analysis, severe accident research, and severe accident analysis. Since PRA considerations have been integrated into the AP1000 design process from the beginning, many of the traditional PRA insights relating to current operating plants are not at issue for the AP1000. The Level 1, Level 2, and Level 3 results show that addressing PRA issues in the design process leads to a low level of risk. The PRA results indicate that the AP1000 design meets the higher expectations and goals for new generation passive pressurized water reactors (PWRs).

The core damage frequency (CDF) and large release frequency (LRF) for at-power internal events (excluding seismic, fire, and flood events) are 2.41E-07 events per reactor-year and 1.95E-08 events per reactor-year, respectively. These frequencies are at least two orders of magnitude less than a typical pressurized water reactor plant currently in operation. This reduction in risk is due to many plant design features, with the dominant reduction coming from highly reliable and redundant passive safety-related systems that impact both at-power and shutdown risks. These passive systems are much less dependent on operator action and support systems than plant systems in current operating plants.

The Level 3 analysis shows the potential offsite dose from a severe accident is very small and well within the established goals. The risk measured by the potential offsite dose does not increase significantly after the first 24 hours after a severe accident is assumed to cause a release to the environment.

Conservative, bounding fire and flood assessments show the core damage risk from these events is small compared to the core damage risk from at-power and shutdown events.

A synopsis of the insights gained from the PRA about the AP1000 design includes:

- The AP1000 design benefits from the high level of redundancy and diversity of the passive safety-related systems. The passive systems have been shown to be highly reliable, their designs are simple so that a limited number of components are required to function.

- AP1000 is less dependent on nonsafety-related systems than current plants or advanced light water reactor evolutionary plants.

- The nonsafety-related support systems (ac power, component cooling water, service water, and instrument air) have a limited role in the plant risk profile because the passive safety-related systems do not require cooling water or ac power.

- AP1000 is less dependent on human actions than current plants or advanced light water reactor evolutionary plants. Even when no credit is taken for operator actions, the AP1000 meets the NRC safety goal, whereas current plants may not.

- The core damage and large release frequencies are low despite the conservative assumptions made in specifying success criteria for the passive systems. The success criteria have been developed in a more systematic, rigorous manner than typical PRA success criteria. The baseline success criteria are bounding cases for a large number of PRA success sequences. The baseline success sequences, in most cases, have been defined with:

  - Worst (i.e., the most limiting) break size and location for a given initiating event

  - Worst automatic depressurization system (ADS) assumption in the success criterion

  - Worst number of core makeup tanks (CMT) and accumulators

  - Worst containment conditions for in-containment refueling water storage tank (IRWST) gravity injection

  Many less-limiting sequences are therefore represented by a baseline success criterion.

- Single system or component failures are not overly important due to the redundancy and diversity of safety-related systems in the design. For example, the following lines of defense are available for reactor coolant system (RCS) makeup:

  - Chemical and volume control system (CVS)

  - Core makeup tanks

  - Partial automatic depressurization system in combination with normal residual heat removal

  - Full automatic depressurization system with accumulators and in-containment refueling water storage tank

  - Full automatic depressurization system with core makeup tanks and in-containment refueling water storage tank

- Typical current PRA dominant initiating events are significantly less important for the AP1000. For example, the reactor coolant pump (RCP) seal loss-of-coolant accident (LOCA) event has been eliminated as a core damage initiator since AP1000 uses canned motor reactor coolant pumps which do not have seals. Another example is the loss of offsite power (LOOP) event. The station blackout and loss of offsite power event is a minor contributor to AP1000 since the passive safety-related systems do not require the support of ac power.

- Passive safety-related systems are available in all shutdown modes. Planned maintenance of passive features is only performed during shutdown modes when that feature is not risk important. In addition, planned maintenance of nonsafety-related defense-in-depth features used during shutdown is performed at power.
- The AP1000 passive containment cooling design is highly robust. Air cooling alone is significant and may prevent containment failure, although the design has other lines of defense for containment cooling such as fan coolers and passive containment cooling water.
- The potential for containment isolation and containment bypass is lessened by having fewer penetrations to allow fission product release. In addition, normally open and risk important penetrations are fail-closed, thus eliminating the dependence on instrumentation and control (I&C) and batteries.
- The reactor vessel lower head has no vessel penetrations, thus eliminating penetration failure as a potential vessel failure mode. Preventing the relocation of molten core debris to the containment eliminates the occurrence of several severe accident phenomena, such as ex-vessel fuel-coolant interactions and core-concrete interaction, which may threaten the containment integrity. Therefore, AP1000, through the prevention of core debris relocation to the containment, significantly reduces the likelihood of containment failure.
- The potential for the spreading of fires and floods to safety-related equipment is significantly reduced by the AP1000 layout.

59.2 Use of PRA in the Design Process

The AP1000 design has evolved over a period of years, including the work done for the AP600 design. PRA techniques have been used since the beginning in an iterative process to optimize the AP600/AP1000 with respect to public safety. Each of these iterations has included:

- Development of a PRA model
- Use of the model to identify weaknesses
- Quantification of PRA benefits of alternate designs and operational strategies
- Adoption of selected design and operational improvements.

The scope and detail of the PRA model has increased from the early studies as the plant design has matured. This iterative design process has resulted in a number of design and operational improvements.

59.3 Core Damage Frequency from Internal Initiating Events at Power

Internal initiating events are transient and accident initiators that are caused by plant system, component, or operator failures. External initiating events, which include internal fire and flooding events and events at shutdown are discussed in other subsections.

The AP1000 mean plant core damage frequency for internal initiating events at power is calculated to be 2.41E-07 events per year. Twenty-six separate initiating event categories were defined to accurately represent the AP1000 design. Of these event categories, 11 are loss-of-coolant accidents, 12 are transients, and 3 are anticipated transients without scram precursors (initiating events that result in an anticipated transient without scram sequence as a result of failure to trip the reactor). Initiating event categories unique to the AP1000 design have been defined and evaluated, including safety injection line breaks, core makeup tank line breaks, and passive residual heat removal heat exchanger (HX) tube ruptures. The resulting core damage frequency is very small; a value of 2.41E-07 means that only one core damage event is expected in 4 million plant-years of operation. This core damage frequency value is two orders of magnitude (i.e., 100 times) smaller than corresponding values typically calculated for current pressurized water reactors.

The contribution of initiating events to the total plant core damage frequency is summarized in Table 59-1. Figure 59-1 illustrates the relative contributions to core damage frequency from the various at-power initiating events. Table 59-2 shows the conditional core damage probability of the initiating events. The conditional core damage probability listed in Table 59-2 is the ratio of the core damage frequency contribution for an initiating event divided by the initiating event frequency.

Seven initiating events, including 6 loss-of-coolant accidents, and steam generator tube rupture (SGTR), make up approximately 92 percent of the total at-power plant core damage frequency. The remaining initiating events contribute a total of approximately 8 percent to the core damage frequency from internal events. The dominant initiating events are:

- Safety injection (DVI) line break
- Large loss-of-coolant accident
- Spurious ADS actuation
- Small loss-of-coolant accident
- Medium loss-of-coolant accident
- Reactor vessel rupture
- Steam generator tube rupture

Within this group of events, each of the first three contribute more than 10 percent to the total core damage frequency. These three events account for approximately 70 percent of the total core damage frequency. Small LOCA, medium LOCA, and reactor vessel rupture events contribute 7 percent, 6 percent and 4 percent, respectively.

The results show a very low core damage frequency dominated by rare events (initiating events that are not expected to occur during the lifetime of a plant). This indicates that the AP1000 design is robust with respect to its ability to withstand challenges from more frequent events (e.g., transients) and that adequate protection against the more severe events is provided through the defense-in-depth features.

Information regarding loss-of-coolant accident categories defined for the AP1000 PRA was presented in the discussion of PRA success criteria. For the PRA, the various loss-of-coolant accident categories have been defined based on which plant features are required to mitigate the events. As a result, the PRA and loss-of-coolant accident size definitions are not identical to the loss of coolant accident size definitions used in the Chapter 15, Accident Analyses included in the *AP1000 Design Control Document* (DCD). The following listing shows how the PRA and DCD break sizes are related and identifies the PRA size criteria:

- DCD Chapter 15 break size definitions are large (break size greater than 1 $ft.^2$) or small (break size less than 1 $ft.^2$).

- PRA break sizes are defined as follows:

  - Large breaks are those with an equivalent inside diameter of approximately 9 in. or larger. Reactor vessel rupture is included in this category. The automatic depressurization system is not required for in-containment refueling water storage tank injection for large breaks. (For large breaks that are slightly larger than a medium break, there is a potential effect of containment isolation upon in-containment refueling water storage tank injection. The success criteria include automatic depressurization system in these cases.)

  - Medium breaks are those with an equivalent inside diameter between approximately 2 in. and 9 in. Core makeup tank line breaks and safety injection line breaks are included in this category (but are evaluated separately). Operation of automatic depressurization system stages 1, 2, or 3 (or, alternatively, passive residual heat removal) is not required to satisfy the automatic depressurization system stage 4 automatic actuation pressure interlock, but is required to depressurize the reactor coolant system to the normal residual heat removal system operating pressure.

  - Small breaks are those with an equivalent inside diameter between approximately 3/8 in. and 2 in. Steam generator tube rupture and passive residual heat removal heat exchanger tube rupture break sizes fall within this range, but are evaluated as separate events based on differing initial plant response. Small breaks are larger than those for which the chemical and volume control system can maintain reactor coolant system water level, but not large enough to allow automatic actuation of automatic depressurization system stage 4 without operation of either automatic depressurization system stages 1, 2, or 3 or passive residual heat removal.

  - Coolant losses smaller than those resulting from small breaks are defined as reactor coolant system leaks. Operation of one chemical and volume control system makeup pump can maintain reactor coolant system water inventory for reactor coolant system leaks.

59.3.1 Dominant Core Damage Sequences

A total of 791 potential core damage event sequences for internal initiating events at power are modeled in the AP1000 PRA. These core damage sequences are the combinations of initiating event occurrences and subsequent successes and failures of plant systems and operator actions that result in core damage. Of these 791 event sequences, 190 result in frequencies ranging from 7-08 to 1E-15 events per year. The remaining sequences do not produce any cutsets representing them in the top 19,000 cutsets; that is, their core damage frequencies are not significant relative to the core damage frequencies for the other sequences.

- The 10 sequences with the highest core damage frequencies together contribute 79 percent of the total (approximately 1.92E-07 events per year).
- The top 19 sequences contribute 90 percent of the total (approximately 2.18E-07 events per year).
- The top 58 sequences contribute 99 percent of the total (approximately 2.39E-07 events per year).
- The top 100 sequences contribute 99.9 percent of the total (approximately 2.41E-07 events per year).

The 19 dominant sequences are given in Table 59-3.

Moreover, each core damage sequence is composed of component-level cutsets, with a total of approximately 19,000 cutsets included in the baseline internal initiating events at-power analysis (100 percent of 2.41E-07 events per year core damage frequency). A cutset is a combination of initiating event occurrence and the component or operator failures that constitute the various system-level failures that lead to core damage.

- The 100 highest-frequency cutsets together contribute approximately 86 percent of the total core damage frequency (approximately 2.1E-07 events per year).
- The top 200 cutsets contribute approximately 91 percent (2.2E-07 events per year). These cutsets are reported in Section 36.
- The top 500 cutsets contribute approximately 95 percent (2.3E-07 events per year).
- The top 1,000 cutsets contribute approximately 97 percent (2.35E-07 events per year).
- The top 2,000 cutsets contribute approximately 98 percent (2.37E-07 events per year).

The top 10 accident sequences contribute 79 percent of the core damage frequency from internal initiating events at power. These sequences are listed in Table 59-3. The top 25 cutsets for these sequences are given in Tables 59-4 through 59-13.

The first four dominant accident sequences make up 63 percent of the core damage frequency. These sequences are:

1. Safety injection line break event occurs, which is postulated to lead to spilling of one train of core makeup tank, in-containment refueling water storage tank, and recirculation flows. The reactor is tripped. The second core makeup tank successfully injects, and the automatic depressurization system is successfully actuated. Thus, the reactor coolant system pressure is low. However, the remaining in-containment refueling water storage tank line fails to inject; core damage occurs with low reactor coolant system pressure, leading to a postulated 3BE end state. The sequence frequency is 6.9E-08 per year, contributing 29 percent to the plant core damage frequency.

2. Large loss-of-coolant accident event occurs, and the reactor is tripped or is rendered subcritical because of voids in the reactor coolant system. Reactor coolant system rapidly depressurizes but one of the accumulators does not inject water into the RCS. Core damage with low reactor coolant system pressure, leading to the 3BR end state is postulated. The sequence frequency is 4.3E-08 per year, contributing 18 percent to the plant core damage frequency.

3. Spurious ADS actuation event occurs, and the reactor is tripped or is rendered subcritical because of voids in the reactor coolant system. Reactor coolant system rapidly depressurizes and at least one of the two accumulators injects, making up the RCS water loss in the short time frame. The CMT injection or ADS actuation fails. Thus, automatic IRWST injection is not actuated. Core damage with medium reactor coolant system pressure, leading to the 3D end state is postulated. The sequence frequency is 2.1E-08 per year, contributing 9 percent to the plant core damage frequency.

4. Safety injection line break event occurs, which is postulated to lead to spilling of one train of core makeup tank, in-containment refueling water storage tank, and recirculation flows. The reactor is tripped. The second core makeup tank successfully injects, but the automatic depressurization system actuation fails. Core damage is postulated with a medium reactor coolant system pressure, leading to a 3D end state. The sequence frequency is 2.0E-08 per year, contributing 8 percent to the plant core damage frequency.

The fifth dominant sequence, with 4 percent contribution to plant core damage frequency, is a reactor vessel rupture event. By the definition of this event, core damage is postulated to occur. The end state is 3C.

59.3.2 Component Importances for At-Power Core Damage Frequency

Chapter 50 presents tables of the relative importances of all basic events appearing in the cutsets for the baseline core damage quantification. These tables indicate risk decrease and risk increase. Risk decrease is the factor by which the core damage frequency would decrease if the failure probability for a given basic event is set to 0.0; it is a useful measure of the benefit that might be obtained as a result of improved component maintenance or testing, better procedures, or operator training. Risk increase is the factor by which the core damage frequency would increase if the failure probability for a given basic event is set to 1.0; it is a useful measure of which components or actions would most adversely affect the core damage frequency if actual operating practices resulted in higher failure probabilities than assumed in the PRA.

The risk decrease results (as discussed in detail in Chapter 50) show that only six components have a risk reduction worth (RRW) of greater than or equal to 1.05. The in-containment refueling water storage tank discharge line strainer plugging has the highest RRW value, followed by common cause failure (CCF) of various components as shown in the following table.

IWA-PLUG	1.27	IRWST discharge Line "A" strainer plugged
ADX-EV-SA2	1.11	CCF of 2 squib valves to operate
REX-FL-GP	1.08	CCF plugging of both recirculation lines due to sump screens
ADX-EV-SA	1.05	CCF of 4th stage ADS squib valves to operate
IWX-CV-AO	1.05	CCF of 4 gravity injection check valves
IWX-EV-SA	1.05	CCF of 4 gravity injection & 2 recirculation squib valves

The remaining components each have a risk reduction worth of 1.04 or less. The contribution to the core damage frequency from unscheduled maintenance is also small. These results indicate that there are no components for which an improvement in design, test, or maintenance (i.e., a change resulting in a significant reduction of the component failure rate) would have a significant impact on the core damage frequency.

Excluding common cause failures, the risk increase results indicate that the accumulator system components have high risk achievement worth (RAW) values, followed by one Non-Class 1E dc and uninterruptible power supply system (EDS) bus, various Class 1E dc and uninterruptible power supply system (IDS) components and CMT components. Other single-component failures have significantly lower risk increase values, corresponding to a factor of six or lower increase in core damage frequency given an assumption of total unreliability for these components.

59.3.3 System Importances for At-Power Core Damage

System importances for plant core damage frequency from internal initiating events at power are presented in Chapter 50. They are obtained by setting the failure probabilities for the affected system components to 1.0 in the baseline cutsets and recalculating the core damage frequency.

The results of the sensitivity analyses show that the protection and safety monitoring system and the Class 1E dc power system are most important in maintaining a low core damage frequency. The risk-important systems are safety-related systems. The safety-related systems are all of high or medium importance. The nonsafety-related systems are only marginally important to the plant core damage frequency.

A sensitivity analysis is made for the unavailability of all five of the standby non-safety related systems (chemical and volume control system (CVS), startup feedwater system (SFW), normal residual heat removal system (RNS), diverse actuation system (DAS), diesel generators (DGs)). The plant CDF obtained is 7.40E-6, which is a factor of 31 increase over the base case. This sensitivity analysis shows that the plant CDF is somewhat sensitive to the simultaneous failure of the five systems listed above.

59.3.4 System Failure Probabilities for At-Power Core Damage

Some selected system failure probabilities for typical success criteria used in the at-power PRA are listed in Table 59-14. A system may have different failure probabilities based on the success criteria assigned. For a key safety-related system such as the automatic depressurization system, this is especially pronounced; the automatic depressurization system has many success criteria and corresponding failure probabilities that range over a factor of 100. The values in the table are representative of the various cases.

As can be seen from the system unavailabilities listed in Table 59-14, the highest unavailabilities (i.e., 10^{-2} to 10^{-3}, indicating lower reliability) are associated with nonsafety-related systems or functions. The lower unavailabilities (i.e., 10^{4} to 10^{-6}, indicating higher reliability) are associated with safety-related systems.

59.3.5 Common Cause Failure Importances for At-Power Core Damage

The common cause importance results are presented in Chapter 50. The risk increase importances for common cause failures of the following sets of components show that these are also of potential significance to the current low level of core damage frequency from internal events: common cause failure of software in the protection and safety monitoring system and plant control system, logic board failures of the protection and safety monitoring system; failures of transmitters used in the protection and safety monitoring system; failures of reactor trip breakers; plugging of containment sump recirculation screens; failures of in-containment refueling water storage tank gravity injection line check valves and squib valves; plugging of strainers in the in-containment refueling water storage tank; failures of fourth-stage automatic depressurization system squib valves and failures of output cards for the protection and safety monitoring system. These and similar common cause failures are of potential significance in maintaining the current level of low plant core damage frequency.

The leading risk decrease common cause failures of hardware are associated with ADS fourth stage squib valves, gravity injection and recirculation line components, and I&C components and sensors.

59.3.6 Human Error Importances for At-Power Core Damage

In the PRA, credit is taken for various tasks to be performed in the control room by the trained operators. These tasks are rule-based and proceduralized. Although these tasks are usually termed operator actions, the tasks almost always refer to the completion of a well-defined mission by trained operators following procedures. Further, not every individual or group error during a mission necessarily fails the mission, since procedural recovery is built into the emergency procedures. Moreover, a very strong diversity is introduced through monitoring of the emergency procedure status trees by a shift technical advisor. These considerations are factored into the PRA evaluation of human errors.

The risk decrease results for operator actions (discussed in Chapter 50) show that there are 10 human actions with importances greater than 1 percent. There are no actions for which the internal initiating events at-power core damage frequency contribution would decrease by more than 3 percent if it were assumed that the operators always were successful. This indicates that there would be no significant benefit from additional refinement of the actions modeled, nor from special emphasis on operator training in these actions (versus other emergency actions).

The risk increase results show that there are only 7 operator actions with importance greater than 100 percent; i.e., these are the only modeled operator actions whose guaranteed failure would result in a core damage increase greater than the base case core damage frequency. The most important action in this ranking (operator fails to diagnose a steam generator tube rupture event) has a risk achievement worth of 6.3. It is followed by manual actuation of ADS with a RAW value of 4.25. These results indicate that the plant design is not overly sensitive to failure of operator actions and the core damage models do not take undue credit for operator response.

A sensitivity analysis was performed in which the failure probabilities for the 30 operator actions are set to 0.0 (perfect operator). The resulting core damage frequency is only slightly smaller. This indicates that perfection in human error probabilities is not risk important at the level of plant risk obtained by the base case; there is no significant benefit to be gained by improving operator response beyond the assumptions made in the PRA.

Another sensitivity analysis was performed in which the failure probabilities for the 30 human error probabilities and also for indication failure (protection and safety monitoring system, plant control system, or diverse actuation system originated) are set to 1.0 (failure). The result of the sensitivity analysis shows that the core damage frequency increased to 1.4E-05 events per year. The resulting core damage frequency with no credit for operator actions is still low (about one event in 71,000 reactor-years), on the order of core damage frequency for current plants with credit for operators. This means that, in general, operator actions are important in maintaining a very low plant core damage frequency for internal events at power but are not essential to establishing the acceptability of plant risk. The presence of trained operators will help ensure that the very low core damage frequency prediction is valid. This finding demonstrates a significantly lower dependence on human actions than exists for current plants. The AP1000 meets the core damage frequency safety goal without human action, whereas current plants typically do not.

59.3.7 Accident Class Importances

The accident classes (also referred to as end states) are described in Chapter 44, and the contribution of accident classes to plant core damage frequency is presented in the same chapter. Two low-pressure reactor coolant system core damage end states, 3BE and 3BL, contribute 43 percent to the total core damage frequency. Together with 3BR and 3D, full or partially depressurized core damage states make up 87 percent of the core damage. In these end states, the probability of retaining containment integrity is very likely. Thus, severe release potential for these end states is low.

59.3.8 Sensitivity Analyses Summary for At-Power Core Damage

Thirty-six importance and sensitivity analyses were performed on the core damage model for internal initiating events at power. These cases and results are discussed in Chapter 50.

The analyses were chosen to address the following issues:

- Importances of individual basic events and their effect on plant core damage frequency
- Importances of safety-related and nonsafety-related systems in maintaining a low plant core damage frequency
- Importances of containment safeguards systems in maintaining a low large-release frequency
- Effect of human reliabilities as a group on plant core damage frequency
- Other specific issues such as passive system check valve reliability, etc.

The sensitivity analyses results are discussed in Chapter 50. They show that:

- If no credit is taken for operator actions, the plant core damage frequency is 1.4E-05 events per year. This compares well with core damage frequencies for existing plants where credit is taken for operator actions.
- The most important systems for core damage prevention are the protection and safety monitoring system, Class 1E dc power, automatic depressurization system, in-containment refueling water storage tank recirculation, core makeup tanks, and accumulators. None of the nonsafety-related systems have high system importance.
- There are no operator actions that would provide a significant risk decrease if they were made to be more reliable. There are only eight operator actions that would increase the core damage frequency by more than the base case if they were assumed to fail. The most important of these is the failure to diagnose a steam generator tube rupture event.
- If the reliability of all check valves is assumed to be a factor of 10 worse, the total plant core damage frequency would only increase to 8.8E-7 events per year. This shows that the passive safety-related systems that depend on check valve opening will perform acceptably, even if pessimistic check valve reliabilities are assumed.
- The plant core damage frequency is not affected by the diesel generator mission time duration. This is due to the AP1000 design's passive features, which do not require ac power for operation.

- The common cause failure basic events, particularly those associated with safety-related systems, are important individually, and also as a group for plant core damage frequency. This is expected for a plant with highly redundant safety-related systems, for which individual component random failure contributions are of reduced significance.

59.3.9 Summary of Important Level 1 At-Power Results

The results of the PRA show that the following AP1000 design features provide the ability to respond to internal initiating events and contribute to a very low core damage frequency:

- The manual feed and bleed operation in current pressurized water reactors is replaced by the automatic depressurization system and core makeup tank/in-containment refueling water storage tank injection. This increases the success probability for feed and bleed and helps reduce core damage contribution from transients with failure of decay heat removal.

- The switchover-to-recirculation operation in current pressurized water reactors is replaced with automatic recirculation of sump water into the reactor coolant system loops by natural circulation.

- The diverse actuation system provides diverse backup for automatic or manual actuation of safety-related systems, increasing the system reliability for the passive residual heat removal, core makeup tank, and automatic depressurization systems.

- The AP1000 plant design is based on a defense-in-depth concept. There are several means (both active and passive) of providing reactor coolant system makeup following a loss-of-coolant accident, at both high and low pressures (i.e., chemical and volume control system pumps, core makeup tanks, accumulators, in-containment refueling water storage tank gravity injection, and normal residual heat removal system). Similarly, there are diverse means of core cooling, including the passive residual heat removal and normal residual heat removal systems.

- The ability to depressurize and establish feed and bleed heat removal via the automatic depressurization system and core makeup tanks without operator action provides an additional reliable means of core cooling and inventory control.

- The diversity and redundancy in the design of the automatic depressurization system provide a highly reliable system for depressurizing to allow injection and core cooling by the various sources of water.

- The design of the reactor coolant pumps eliminates the dependence on component cooling water and accompanying reactor coolant pump seal loss-of-coolant accident core damage contribution, which is typically significant for current plants.

- The design of the safety-related heat removal systems eliminates the dependence on service water and ac power during accidents; such dependencies can be significant contributors to core damage for current plants.

Core Damage Contribution from Important Initiating Events

Loss-of-Coolant Events. The at-power core damage results are dominated (top 8 dominant contributors with 93 percent) by various loss-of-coolant events. Thirty-four percent of the contribution is due to the safety injection line break, which is a special initiator, in that its occurrence partially defeats features incorporated into the plant to respond to losses of primary coolant. Even though the safety injection line break core damage frequency dominates the results, its value is very small (one event in 10 million reactor years), with little credit for nonsafety-related systems.

The conditional probability of core damage, given the occurrence of a "conventional" loss-of-coolant accident, is generally in the range of about 1E-03 to 1E-05 (with the exception of reactor vessel rupture and interfacing systems loss-of-coolant accident, for which core damage is assumed). These events have frequencies of about 1E-08 per year to 5E-04 per year. This indicates that the various features of the AP1000 would act to prevent core damage from all but between 1 in 1000 and 1 in 100,000 loss-of-coolant accidents. Since loss-of-coolant accidents are relatively rare events, this is a significant level of protection.

Anticipated Transients Without Scram. Anticipated transients without scram (ATWS) sequences contribute about 2 percent of the at-power core damage frequency, in part due to modeling simplifications whereby, in the absence of specific modeling and success criteria, it has been assumed that core damage will occur given certain combinations of failures. With additional analysis and modeling detail, it is expected that the anticipated transient without scram core damage frequency could be shown to be lower.

Transients. The contribution of transients to core damage frequency is about 5 percent of the at-power core damage frequency (total contribution from all transient initiators with reactor trip is 1 event in 100 million reactor years). This is the result of the defense-in-depth features of the AP1000 design, whereby core cooling following transients is available from main feedwater, startup feedwater, and passive residual heat removal, as well as from feed and bleed, using diverse and redundant sources of makeup (core makeup tanks, accumulators, in-containment refueling water storage tank, normal residual heat removal system), and of depressurization (four stages of automatic depressurization system).

Loss of Offsite Power. The loss of offsite power core damage frequency contribution at power is insignificant (less than 1 percent). AP1000 passive systems require only dc power provided by the long-term batteries for actuation to provide cooling. In addition, the passive residual heat removal heat exchanger is backed up by bleed and feed cooling using the automatic depressurization system and core makeup tanks or in-containment refueling water storage tank gravity injection, which also require only dc power provided by long-term batteries. With onsite power available, startup feedwater provides an additional means of decay heat removal.

Steam Generator Tube Rupture. The steam generator tube rupture event contributes about 3 percent of the at-power core damage frequency. Compared to operating pressurized water reactors this is a very low contribution. Among the reasons for the small steam generator tube rupture core damage contribution are the following:

- The first line of defense is the startup feedwater system and chemical and volume control system
- A reliable safety-related passive residual heat removal system coupled with the core makeup tank subsystem, which provides automatic protection
- A third line of defense using automatic depressurization system and in-containment refueling water storage tank for accident mitigation should the above-mentioned systems fail.

Further, the automatic depressurization system provides a more reliable alternate decay heat removal path through feed and bleed than the high-pressure manual feed and bleed cooling of current operating plants.

Finally, the large capacity of the in-containment refueling water storage tank increases the long-term recovery probability for unisolable steam generator leaks that bypass containment, by preventing depletion of borated water and core damage.

Dependence on Operator Action

The results of the PRA show that the AP1000 is significantly less dependent on operator action to reduce plant risk to acceptable levels than are current plants. This was shown through the sensitivity analyses and the operator action contributions from both the risk decrease and risk increase measures. Almost all operator actions credited in this PRA are performed in the control room; there are very few local actions outside the control room. Further, the human actions modeled in the AP1000 PRA are generally simpler than those for current plants. Thus, the tasks for AP1000 operators are easier and less likely to fail. If it were assumed that the operators never perform any actions credited in the PRA, the internal events core damage frequency would still be lower than the result obtained for many current pressurized water reactors including operator actions.

Dominant System/Component Failure Contributors

Contribution to Core Damage Frequency. Component-related contributors to core damage frequency from internal events at power are dominated by common cause failures. The single component failures are limited to strainer or tank failures, and accumulator check valve failures.

Dependence on Component Reliability. Most of the component failures with relatively high risk increase worth are common cause failures. This is an indication of the high degree of built-in redundancy and diversity of AP1000 safety-related systems, particularly in view of the low baseline core damage frequency. The results demonstrate a well-balanced design, for which diversity eliminates the strong dependence on active valves or on the specific type of valve.

Sensitivity to Numerical Values and Modeling Assumptions. The core damage results are not strongly sensitive to increases in the failure probabilities of basic events. Check valves are relatively important; if the check valve failure probability is increased by a factor of 10, the core damage frequency increases by a factor of 4. This increase is not large, and the core damage goal of 1E-05 is comfortably met. Finally, the modeling assumptions in system and accident sequence success criteria are bounding (e.g., conservative) whenever a range of conditions are represented by a single selected condition or success criterion. Since the modeling assumptions already represent an upper bound type estimate, there are no significant contributions to core damage due to conditions outside the assumed ranges that are unaccounted for. As an example, the automatic depressurization system success criteria for loss-of-coolant accident events are selected to cover the worst conditions (e.g., break size, break location) of the range.

System Reliability and Defense-in-Depth. The results show that the safety-related systems have demonstrated high reliabilities (e.g., failure probability in the range of 1E-05 to 1E-03), due to the nature of the system designs (passive systems). Moreover, multiple means of success exist for transients and credible loss-of-coolant accident events. This means that a failure of a safety-related system will not lead to core damage, because other diverse systems back up the first one. This defense-in-depth philosophy contributes to the low core damage frequency.

59.4 Large Release Frequency for Internal Initiating Events at Power

The results of the Level 2 (containment response) and Level 3 (plant risk) analyses for the internal initiating events at power demonstrate that the AP1000 containment design is robust in its ability to prevent releases following a severe accident and that the risk to the public due to severe accidents for AP1000 is very low. The large release frequency (containment failure frequency) of the AP1000 can be divided into two types of failures: 1) initially failed containment, in which the integrity of the containment is either failed due to the initiating event or never achieved from the beginning of the accident; and 2) containment failure induced by high-energy severe accident phenomena. The total of these failures is the overall large release frequency. The following summarizes important results of the containment event tree quantification with respect to large release frequency.

The overall release frequency for AP1000 is 1.95E-08 events per year. This is approximately 8 percent of the core damage frequency for internal initiating events at power. The ability of the containment to prevent releases (i.e., the containment effectiveness) is 92 percent.

The Level 3 analysis shows that the resulting risk to the population is small and well within the established goals.

59.4.1 Dominant Large Release Frequency Sequences

The large release frequency is dominated by release categories BP (bypass), with a 54-percent contribution and CFE (early containment failure) with a contribution of 38 percent. The total frequency of these two categories is 1.8E-08 events per year. These two categories make up 92 percent of the plant large release frequency, followed by 7.0 percent contribution from containment isolation failure category. Contributions of the late containment failure (CFL) and intermediate containment failure (CFI) release categories to large release frequency are negligible.

The early containment failures are caused by sump flooding, vessel failure, and core reflooding failure plus containment overtemperature failure due to diffusion flame.

The dominant accident class in the large release frequency is the Class 6 with a 21-percent contribution. This class represents sequences in which steam generator tube rupture or interfacing LOCA events occur. It is followed by accident class 3A, with a 21 percent contribution. 3A contains core damage events with high RCS pressure and ATWS events.

The dominant large release frequency sequences are shown below. These sequences make up 98 percent of the large release frequency. Two containment bypass sequences from 3A and 6 accident classes contribute 21 percent and 19 percent, followed by 2 early containment failures from 3BE and 3D accident sequences with 14 and 11 percent contributions. These four sequences add up to 65 percent of the plant LRF.

Dominant Containment Event Tree (CET) Sequences					
CET SEQ	REL CAT	PDS	FREQ	%	SEQUENCE DESCRIPTION
23	BP	3A	4 08E-09	20.9%	Containment Bypass
23	BP	6	3.78E-09	19.4%	Containment Bypass
21	CFE	2E	2.67E-09	13.7%	Sump Flooding Fails
21	CFE	3D	2.05E-09	10.5%	Sump Flooding Fails
23	BP	1A	2.04E-09	10.5%	Containment Bypass
10	CFE	3C	9.97E-10	5.1%	Vessel Failure
12	CFE	3D	9.71E-10	5.0%	Core Reflooding Fails; Diffusion Flame
23	BP	1P	6.05E-10	3.1%	Containment Bypass
22	CI	2L	5.83E-10	3.0%	Containment Isolation Fails
6	CFE	2E	4.75E-10	2.4%	Hydrogen Igniters Fail; Early deflagration to detonation transition (DDT)
22	CI	3D	3.62E-10	1.9%	Containment Isolation Fails
21	CFE	6	1.86E-10	1.0%	Sump Flooding Fails
4	CFI	2E	1.82E-10	0.9%	Hydrogen Igniters fail; Intermediate DDT

59.4.2 Summary of Important Level 2 At-Power Results

The results of the PRA show that the following AP1000 design features provide the ability to respond to various severe accidents and contribute to a very small release frequency and a small release of radioactive material to the environment.

- The capability to flood the reactor cavity prevents the failure of the reactor vessel given a severe accident without water in the cavity. The vessel and its insulation are designed so that the water in the cavity is able to cool the vessel and prevent it from failing (in-vessel retention - IVR). By maintaining the vessel integrity, the core debris in the vessel eliminates the potential of a large release due to ex-vessel phenomena and its potential to fail the containment.

- The capability to depressurize the reactor coolant system in a high-pressure transient mitigates the consequences of a high-pressure severe accident. Such accidents have a large potential to fail the reactor coolant system pressure boundary vessel, piping, or steam generator tubes, and such a failure is assumed without further analysis if the reactor coolant system remains at high pressure. A high-pressure failure of the reactor coolant system pressure boundary is assumed to fail or bypass the containment. Thus, the capability to depressurize the reactor coolant system reduces the large release frequency due to high-pressure severe accidents.

- The annular spaces between the steel containment vessel and the shield building help to reduce the release of radioactive materials to the environment by enhancing the deposition of the materials before they exit the containment.

The Level 2 results highlight some insights in the AP1000 design:

- The containment effectiveness for AP1000 is over 90 percent, which provides an order of magnitude decrease from CDF to LRF. Since this result already includes CDF sequences that directly bypass the containment, the containment effectiveness for remaining sequences is actually much better. For example, for 5 (3BE, 3BL, 3BR, 3C, 3D) of the 9 accident classes studied, the containment effectiveness ranges from 90 to 99.8 percent.

- The containment effectiveness is lowest for the 3A accident class where the RCS pressure is high after core damage. The post-core-damage depressurization for this class proves to be ineffective since failure of ADS by common cause failures leading to core damage also causes failure of post-core-damage depressurization.

- Based on detailed analysis, the containment effectiveness for accident class 6, mainly SGTR events, is 56.9 percent, due to those sequences where the RCS pressure is low after the postulated core damage. In such sequences, the fission products can be retained in the pressure vessel, shielded by the water in the faulted steam generator. A sensitivity analysis where all accident class 6 events are assigned to LRF shows that the plant containment effectiveness drops slightly to 89.7 percent (from 91.9 percent). Thus, the LRF results are not very sensitive to the treatment of the SGTR events for LRF.

- A frequency of 1.0E-08/year has been assigned to the vessel failure initiating event (accident class 3C). In 90 percent of these events, the vessel is assumed to undergo failures that will be above the beltline – in which case the molten core could be cooled and containment would not be challenged. In the remaining 10 percent of the cases, the failure is assumed to be below the pressure vessel beltline, whereby the molten core would drop into the containment. In this case, it is conservatively assumed that the containment would fail. A sensitivity analysis is made where by 100 percent of the failures would be below the beltline. The result shows that the containment effectiveness drops to 88.2 percent. This change is not significant, and the assumptions behind the case are very conservative.

- The LRF results are sensitive to failure of hydrogen igniters. If no credit is taken for hydrogen igniters, the containment effectiveness drops to 74 percent.

- However, LRF is not very sensitive to the reliability of hydrogen igniters; if IG reliability is assumed to be degraded (0.1) across the board for all accident classes, the containment effectiveness becomes 90.5 percent, which is an insignificant change from the base case.

- For accident classes 3D and 1AP, if the large hydrogen releases through the IRWST is conservatively assumed to cause containment failure, the containment effectiveness drops to 84.5 percent. The LRF increases to 7.58E-08/year. The increase is about a factor of 4 of the base. Such an increase is significant. This sensitivity analysis addresses the uncertainties in hydrogen mixing model for the case where the hydrogen is released into the IRWST and comes out from the IRWST vents above the operating deck.

- The LRF is dominated (53.9 percent) by containment failures or bypasses due to SGTR, and unmitigated high-RCS-pressure core damage sequences, classified as BP. The remaining containment failures are dominated by an early containment failure due to reactor cavity flooding failure.

- The LRF is not very sensitive to the reliability of PCS. If PCS reliability is assumed to be 0.001 across the board for all accident classes, the LRF becomes 1.97E-08, which is an insignificant change from the base case.

- The LRF is sensitive to the operator action to flood the reactor cavity in a short time following core damage. This operator action has been moved to the beginning of Emergency Response Guideline (ERG) AFR.C-1 to increase its likelihood of success.

- The potential for a release of radioactive materials to the environment is very small. This is largely due to the very small core damage frequency and very small release frequency. The containment design provides enhanced deposition of core materials that could be released in a severe accident, and the passive containment cooling system minimizes the energy available to expel such materials from the containment.

The results of the at-power analyses show the AP1000 design includes redundancy and diversity not found in current plants. The safety-related passive systems do not require ac power or operator actions to actuate, and the plant design is robust in the prevention and mitigation of the consequences of an accident. The AP1000 core damage frequency and large release frequency are much lower than has been seen in current generation plants, despite the many conservatisms built into the PRA models. The assumed dose to the environment given a severe accident and a large release is well within the goals set for that analysis.

59.5 Core Damage and Severe Release Frequency from Events at Shutdown

59.5.1 Summary of Shutdown Level 1 Results

As shown by the dominant cutsets of the AP600 and AP1000 shutdown models (shutdown risk evaluation is presented in Chapter 54), the risk profiles of these plants for events during shutdown conditions are almost identical. The results indicate that the three events dominating the CDF are loss of component cooling/service water during drained condition, loss of offsite power during drained condition, and loss of RNS during drained condition. The AP1000 and AP600 initiating event core damage contributions are included in Chapter 54. This data shows the initiating event importance to be similar for the two plants.

The dominant sequences are described in the subsections that follow. The 12 dominant accident sequences comprise 77 percent of the level 1 shutdown core damage frequency. These dominant sequences consist of:

- Loss of component cooling or service water system initiating event during drained condition with a contribution of 64 percent of the CDF
- Loss of RNS initiating event during drained condition with a contribution of 6 percent of the CDF
- Loss of offsite power initiating event during drained condition with a contribution of 5 percent of the CDF
- RCS overdraining event during drainage to mid-loop with a contribution of a 2 percent of the CDF

Loss of Component Cooling or Service Water System Initiating Event During Drained Condition

These sequences are described as the loss of decay heat removal initiated by failure of the component cooling water or service water system during drained condition. The loss of decay heat removal occurs following loss of circulating water system (CWS) or service water system (SWS) during mid-loop/vessel flange operation, which has an estimated duration of 120 hours per 18 months refueling.

The major contributors to risk due to loss of CWS or SWS during drained condition are the following failures:

- Hardware failures of both service water pumps or common cause failure of output logic inputs/outputs (I/Os) from the plant control system (PLS)
- Common cause failure of the ADS 4th stage squib valves
- Common cause failure of the IRWST high-pressure squib valves

- Common cause failure of the strainers in the IRWST tank

- Common cause failure of the recirculation sump strainers

Loss of RNS Initiating Event During Drained Condition

This sequence is described as the loss of decay heat removal initiated by failure of the RNS during drained condition. The loss of decay heat removal occurs following loss of RNS during mid-loop/vessel flange operation, which has an estimated duration of 120 hours per 18 months refueling.

The major contributors to risk due to loss of RNS during drained condition are the following failures:

- Common cause failure of the RNS pumps to run
- Common cause failure of the ADS 4th stage squib valves
- Common cause failure of the IRWST injection squib valves
- Common cause failure of the strainers in the IRWST tank
- Common cause failure of the recirculation sump strainers

Loss of Offsite Power Initiating Event During Drained Condition (with failure of grid recovery within 1 hour)

This sequence is initiated by loss of offsite power during mid-loop/vessel flange operation, which has an estimated duration of 120 hours per 18 months refueling. Following this initiating event, the RNS does not restart automatically, and the grid is not recovered within 1 hour.

The major contributors to risk given loss of offsite power (without grid recovery) are the following failures:

- Software common cause failure of all cards
- Failure of the RNS pump to run or restart
- Failure of the diesel generator to start or run
- Failure of the main breaker to open
- Failure to recover ac power within 1 hour
- Common cause failure of the ADS 4th stage squib valves
- Common cause failure of the IRWST injection squib valves
- Common cause failure of the strainers in the IRWST tank
- Common cause failure of the recirculation sump strainers

Loss of Offsite Power Initiating Event During Drained Condition (with success of grid recovery within 1 hour)

This sequence is initiated by loss of offsite power during mid-loop/vessel flange operation which has an estimated duration of 120 hours per 18 months refueling. Following this initiating event, the RNS does not restart automatically, the grid is recovered within 1 hour but manual RNS restart after grid recovery fails.

The major contributors to risk, given loss of offsite power (with grid recovery), are the following failures:

- Software common cause failure of all cards
- Failure of the RNS pump to run or restart
- Common cause failure of the ADS 4th stage squib valves
- Common cause failure of the IRWST injection squib valves
- Common cause failure of the strainers in the IRWST tank
- Common cause failure of the recirculation sump strainers

RCS Overdraining Event During Drainage to Mid-loop

This sequence is described as RCS overdraining initiating event during drainage to mid-loop condition; draining to mid-loop has an estimated duration of 39 hours per 18 months refueling. Following the initiating event, manual isolation of the RNS fails.

The major contributors to risk due to RCS overdraining are the following failures:

- Common cause failure of the CVS air-operated valves to close automatically upon receipt of low hot leg level signals and failure of the operator to stop draining

- Operator fails to isolate the RNS

- Common cause failure of the ADS 4th stage squib valves

- Operator fails to open IRWST injection squib valves

- Common cause failure of the strainers in the IRWST tank

- Common cause failure of the recirculation sump strainers

Conclusions

The conclusions drawn from the shutdown Level 1 study are as follows:

- The overall shutdown core damage frequency is very small (1.23E-07/year).

- Initiating events during reactor coolant system drained conditions contribute approximately 90 percent of the total shutdown core damage frequency. Loss of decay heat removal capability (during drained condition) due to failure of the component cooling water system or service water system are the initiating events with the greatest contribution (approximately 70 percent of the shutdown core damage frequency).

- Common cause failures of in-containment refueling water storage tank components contribute approximately 59 percent of the total shutdown core damage frequency.

Common cause failure of the in-containment refueling water storage tank valves contributes approximately 33 percent of the total shutdown core damage frequency.

- Common cause failures of the automatic depressurization system stage 4 squib valves contribute approximately 18 percent to the total shutdown core damage frequency. The function of the automatic depressurization system is important to preclude the effects of surge line flooding. This indicates that maintaining the reliability of the automatic depressurization system is important.

- Common cause failures of the containment sump recirculation squib valves contribute approximately 15 percent to the total shutdown core damage frequency. This function is important during drained conditions. This indicates that maintaining the reliability of the recirculation line squib valves is important.

- Human errors are not overly important to shutdown core damage frequency. There is no particular dominant contributor. Sensitivity results show that the shutdown core damage frequency would remain very low even with little credit for operator actions.

  One action, operator failure to recognize the need for reactor coolant system depressurization during safe/cold shutdown conditions, is identified as having a significant risk increase value. This indicates it is important that the procedures include this action and the operators understand and are appropriately trained for it.

- Individual component failures are not significant contributors to shutdown core damage frequency, and there is no particular dominant contributor. This confirms the at-power conclusion that single independent component failures do not have a large impact on core damage frequency for AP1000 and reflects the redundancy and diversity of protection at shutdown as well.

- The in-containment refueling water storage tank provides a significant benefit during shutdown because it serves as a passive backup to the normal residual heat removal system.

59.5.2 Large Release Frequency for Shutdown and Low-Power Events

The baseline PRA shutdown large release frequency for AP600 was calculated to be 1.5E-08 per reactor-year, associated with a shutdown CDF of 9.0E-08 per year. The AP1000 LRF is estimated to be 2.05E-08 per year, with the same risk profile as that of AP600 (see Table 19.59-15). This LRF compares well with the at-power LRF of 1.95E-08 per year.

59.5.3 Shutdown Results Summary

The results of the low-power and shutdown assessment show that the AP1000 design includes redundancy and diversity at shutdown not found in current plants. In particular, the in-containment refueling water storage tank provides a unique safety backup to the normal residual heat removal system. Maintenance at shutdown has less impact on the defense-in-depth features for AP1000 than for current plants. In accordance with plant technical specifications, safety-related system planned maintenance is performed only during those shutdown modes when the protection provided by the safety-related system is not required. Further, maintenance of nonsafety systems, such as the normal residual heat removal system, component cooling water system, and service water system, is performed at power to avoid adversely affecting shutdown risk. These contribute to the extremely low shutdown core damage and the small release frequency.

59.6 Results from Internal Flooding, Internal Fire, and Seismic Margin Analyses

59.6.1 Results of Internal Flooding Assessment

A scoping internal flooding analysis was performed based on AP1000 design information, with conservative assumptions or engineering judgement used for simplifying the analysis.

The AP1000 design philosophy of minimizing the number of potential flooding sources in safety-related areas, along with the physical separation of redundant safety-related components and systems from each other and from nonsafety-related components, minimizes the consequences of internal flooding. The core damage frequencies from flooding events at power is not an appreciable contributor to the overall AP1000 core damage frequency. The internal flooding-induced core damage frequencies are estimated to be 8.8E-10 events per year for power operations.

The internal flooding analysis conservatively assumes that flooding of nonsafety-related equipment results in system failure of the affected system. As shown in AP600 PRA, this results in a higher flooding-induced core damage frequency at shutdown than at power, because of the use of the nonsafety-related normal residual heat removal system as the primary means of decay heat removal at shutdown.

The top five at-power flooding scenarios comprise 91 percent of the at-power flooding-induced core damage frequency. Each of these scenarios relate to large pipe breaks in the turbine building with an initiating event frequency in the range of 1.4 - 2.0E-03/year, leading to a loss of CCS/SWS event. Each scenario has a CDF of 1.2 - 1.8E-10/year.

59.6.2 Results of Internal Fire Assessment

The total at-power, fire-induced core damage frequency, is 5.61E-08 per reactor year. Results of the AP1000 fire PRA analysis are summarized below.

The estimated core damage frequency from main control room fires at power is insignificant (less than 3.18E-12 per year). This low contribution is a result of the following:

- The ignition frequency is low because of the use of low-voltage 48v 10 mA dc cables in the control room. These low-voltage cables do not produce enough energy to heat the cables, thus ignition is not probable.
- Redundancy in control room operations is available within the control room itself; that is, if control room evacuation is not required, there is at least one other means available within the control room to shut down and control the plant.

- If control room evacuation is necessary, the remote shutdown workstation provides complete redundancy in terms of control for safe shutdown functions.

- Loss of control of one division of power or for a whole system is not risk-significant. In addition, the passive systems are designed to operate without the need for operator interaction. Therefore, operator actions that might be disrupted by the fire scenario are backup actions, and are not significant for AP1000.

The results of the internal fire evaluation indicate that the plant's system and layout promote a low fire-induced core damage frequency compared with existing plants. Also, the results indicate that, when nonsafety-related systems are not credited and containment is treated as a special case, the fire-induced core damage frequency profile is relatively flat (i.e., no fire area is significantly more important than others).

The results from the AP1000 fire analysis confirm that the inherent design characteristics of the AP1000 also provide an effective barrier against fire hazards. This is true even within the pessimistic assumptions used throughout the study.

Conservatisms employed in the AP1000 fire analysis included the following:

- In order to minimize potential uncertainty in the results arising from the lack of as-built equipment location and cable routing information, a bounding approach to quantification was taken in accordance with the reference methodology.

- A fire originating from any ignition source in an area is assumed to disable all equipment located in the fire area. The historical evidence indicates that most fires are localized fires with limited severity.

- An assumed total at-power fire initiating event frequency corresponding to about one fire with significant consequences every 4 reactor years, well in excess of current plant experience and of that anticipated for AP1000, was assumed.

- Manual fire suppression is not credited to limit the extent of damage in an area nor to prevent fire propagation to an adjoining area. Historical evidence indicates that the majority of suppressed fires were manually suppressed with little or no additional damage.

- The assumption was made that a single hot short could result in spurious automatic depressurization system actuation.

- The estimation of containment fire frequency, not normally included in fire risk assessments, was done by making a conservative interpretation of the limited available data.

Because the approach taken in performing the internal fire analysis makes various conservative assumptions and is bounding, the results of uncertainty, sensitivity, or importance analyses would be biased. Therefore, these analyses were not performed based on the judgement that they would be of little value in providing additional insights to determine whether fire vulnerabilities exist for beyond-design-basis fires.

The major reasons for the AP1000's relatively low overall fire-induced core damage frequency, even on a bounding basis, include the following:

- The fire protection design provides, to the extent possible, separation of the alternate safety-related shutdown components and cabling using 3-hour-rated fire barriers. For example, areas containing safety-related cabling or components are physically separated from one another and from the areas that do not contain any safety-related equipment by 3-hour-rated fire barriers. This defense-in-depth feature diminishes the probability of a fire to impact more than one safety-related shutdown system.

- Since the passive safety-related systems do not require cooling water or ac power, they are less susceptible to being unavailable due to a fire than currently operating plants' active safe shutdown equipment. As a result, the impact of fires on the shutdown capability is significantly reduced compared to current plants.

The results of this analysis show that the AP1000 design is sufficiently robust that internal fires during either power operation or shutdown do not represent a significant contribution to core damage frequency.

59.6.3 Results of Seismic Margin Analysis

The seismic margin analysis (SMA) shows the systems, structures, and components required for safe shutdown. The high confidence, low probability of failure (HCLPF) values are greater than or equal to 0.50g. This HCLPF is determined by the seismically induced failure of the fuel in the reactor vessel, core assembly failures, IRWST failure, or containment interior failures. The SMA result assumes no credit for operator actions at the 0.50g review level earthquake, and assumes a loss of offsite power for all sequences.

The seismic margin analysis shows the plant to be robust against seismic event sequences that contain station blackout coupled with other seismic or random failures. The analysis also shows the plant's capability to respond to seismic events without benefit of the operators' actions.

59.7 Plant Dose Risk From Release of Fission-Products

Chapter 49 discusses the Level 3 results for at-power and shutdown internal events. The dose risks are quantified by multiplying the fission product release category frequency vector by the release category mean dose vectors. The goal is that a 24-hour, whole-body, site boundary dose greater than 25 rem has a frequency (large release frequency) of less than 1E-06 per year. The AP1000 large release frequency is 1.95E-08 per year, which is a factor of 50 times less than the goal.

The total at-power risk from a postulated release of fission products (the 24-hour, site boundary effective dose equivalent (EDE) is 5.15E-05 rem per reactor-year. For shutdown, this risk was calculated to be 7.1E-05 rem per reactor-year for AP600. For AP1000, this shutdown risk could be estimated as 9.7E-05 rem per reactor-year (estimated the same way as shutdown LRF in Table 59-15). Table 59-16 and Figure 59-2 summarize the plant dose results.

Early containment failures account for 61 percent of the dose risk. These types of failures are usually assumed as a result of sump flooding failure, vessel failure, or core reflooding failure. A less conservative analysis of the early containment failures may show a smaller frequency, and, as a result, a smaller dose risk.

59.8 Overall Plant Risk Results

The total plant risk expressed in terms of plant core damage frequency and severe release frequency for all events studied in this PRA are summarized in Table 59-17.

The contribution of various events to the at-power core damage frequency is shown in Figure 59-1.

The total plant core damage and large release frequency analysis results show the following:

- The total mean core damage frequency is at least two orders of magnitude smaller than those for existing pressurized water reactors. The cumulative core damage probability for a population of 50 AP1000 units operating for 60 years each would be less than 0.001, which is a low probability of occurrence.

- The total plant severe release frequency is another order of magnitude smaller than that of the core damage frequency; that places such a release frequency in the range of incredible events.

- A bounding analysis of the core damage due to internal fire and internal flooding events shows that these two categories of internal events are lower for AP1000 than are calculated for currently operating plants.

- The severe release frequency is about equal for at-power and shutdown events. The severe release frequency as a percentage of core damage frequency is 8 percent for at-power events and 17 percent for shutdown events.

- The results show that the design goals of low core damage frequency and low severe release frequency have been met. The AP1000 frequencies are lower than the Nuclear Regulatory Commission (NRC) goals set for new plant designs, as shown in Table 59-17. These results show the effectiveness of passive systems in mitigating severe accidents and reflect the reduced dependence of AP1000 on nonsafety systems and human actions.

Figure 59-2 shows the 24-hour, whole-body EDE site boundary dose cumulative distribution.

59.9 Plant Features Important to Reducing Risk

Westinghouse used PRA results extensively in the AP1000 design process to identify areas for design improvement and areas for further risk reduction. These results were also compared with existing commercial nuclear power plants to identify additional area of risk reduction. Examples of the more significant AP1000 plant features and operator actions that reduce risk are discussed in this section. Examples are provided in the area of reactor design, system design, plant structures and layout, and containment design.

AP1000 has more lines of defense as compared to current operating plants, which provide more success paths following an initiating event and provide redundancy and diversity to address common cause-related concerns. Examples of extensive AP1000 lines of defense follow:

- Criticality control:
  - Control rod insertion via reactor trip breaker opening
  - Control rod insertion via motor-generator set de-energization
  - Ride out via turbine trip
- Core heat removal:
  - Main feedwater
  - Startup feedwater
  - Passive residual heat removal
  - Automatic depressurization system and feed-and-bleed via normal residual heat removal injection
  - Automatic depressurization system and passive feed-and-bleed via in-containment refueling water storage tank injection
- Reactor coolant system makeup:
  - Chemical and volume control system
  - Core makeup tanks
  - Automatic depressurization system and normal residual heat removal
  - Automatic depressurization system, accumulators, and in-containment refueling water storage tank injection
  - Automatic depressurization system, core makeup tanks, and in-containment refueling water storage tank injection

- Containment cooling:
  - Fan coolers
  - Normal residual heat removal
  - Passive containment cooling system with passive water drain
  - Passive containment cooling system with alternate water supply
  - Passive containment cooling system without water (air only)
  - Fire water

59.9.1 Reactor Design

The AP1000 reactor coolant system has many features that reduce the plant risk profile. The pressurizer is larger than those used in comparable current operating plants, resulting in a longer drainage time during small loss-of-coolant accident events. The larger pressurizer increases transient operation margins, resulting in a more reliable plant with fewer reactor trips, avoiding challenges to the plant and operator during transients. The larger pressurizer also eliminates the need for fast-acting power-operated relief valves (PORVs), which are a possible source of reactor coolant system leaks.

The AP1000 steam generators have large secondary-side water inventories, allowing significant time to recover steam generator feedwater or other means of core heat removal. The AP1000 steam generators also employ improved materials and design features that significantly reduce the probability of forced outages or tube rupture.

The AP1000 has canned reactor coolant pumps, thus avoiding seal loss-of-coolant accident issues and simplifying the chemical and volume control system. The reactor coolant system has fewer welds, which reduces the potential for loss-of-coolant accident events. The probability of a loss-of-coolant accident is also reduced by the application of "leak-before-break" to reactor coolant system piping.

59.9.2 Systems Design

System design aspects that are intended to reduce plant risk are discussed in terms of safety-related and nonsafety-related systems.

59.9.2.1 Safety-Related Systems

The AP1000 uses passive safety-related systems to mitigate design basis accidents and reduce public risk. The passive safety-related systems rely on natural forces such as density differences, gravity, and stored energy to provide water for core and containment cooling. These passive systems do not include active equipment such as pumps. One-time valve alignment of safety-related valves actuates the passive safety-related systems using valve operators such as:

- DC motor-operators with power provided by Class 1E batteries

- Air-operators that reposition to the safeguards position on a loss of the nonsafety-related compressed air that keeps the safety-related equipment in standby

- Squib valves
- Check valves

The passive systems are designed to function with no operator actions for 72 hours following a design basis accident. These systems include the passive containment cooling system and the passive residual heat removal system.

Diversity among the passive systems further reduces the overall plant risk. An example of operational diversity is the option to use passive residual heat removal versus feed-and-bleed for decay heat removal functions, and an example of equipment diversity is the use of different valve operators (motor, air, squib) to avoid common cause failures.

The passive residual heat removal heat exchanger protects the plant against transients that upset the normal steam generator feedwater and steam systems. The passive residual heat removal subsystem of the passive core cooling system contains no pumps and significantly fewer valves than conventional plant auxiliary feedwater systems, thus increasing the reliability of the system. There are fewer potential equipment failures (pumps and valves) and less maintenance activities.

For reactor coolant system water inventory makeup during loss-of-coolant accident events, the passive core cooling system uses three passive sources of water to maintain core cooling through safety injection: the core makeup tanks, accumulators, and in-containment refueling water storage tank. These sources are directly connected to two nozzles on the reactor vessel so that no injection flow can be spilled for larger pipe break events.

The automatic depressurization system is incorporated into the design for depressurization of the reactor coolant system. The automatic depressurization system has 10 paths with diverse valves to avoid common cause failures and is designed for automatic or manual actuation by the protection and safety monitoring system or manual actuation by the diverse actuation system. The automatic depressurization system can be used in a partial depressurization mode to provide long-term reactor coolant system cooling with normal residual heat removal system injection, or it can be used in full depressurization mode for passive in-containment refueling water storage tank injection for long-term reactor coolant system cooling. Switchover from injection to recirculation is automatic without manual actions.

The safety-related Class 1E dc and UPS system has a battery capacity sufficient to support passive safety-related systems for 72 hours. This system has four 24-hour batteries, two 72-hour batteries, and a spare battery. The presence of the spare battery improves testability.

The passive containment cooling system provides the safety-related ultimate heat sink for the plant. Heat is removed from the containment vessel following an accident by a continuous natural circulation flow of air, without any system actuations. By using the passive containment cooling system following an accident, the containment stays well below the predicted failure pressure. The steaming and condensing action of the passive containment cooling system enhances activity removal.

AP1000 containment isolation is significantly improved over that of conventional PWRs due to a large reduction in the number of penetrations. The number of normally open penetrations is reduced. Containment isolation is improved due to the chemical and volume control system being a closed system, the safety-related passive safety injection components being located inside the containment, and the number of heating, ventilation, and air-conditioning (HVAC) penetrations being reduced (no maxi purge connection).

Vessel failure potential upon core damage is reduced (in-vessel retention of the damaged core) by providing a provision to dump in-containment refueling water storage tank water into the reactor cavity. The vessel insulation enables this water to cool the vessel.

For events at shutdown, AP1000 has passive safety-related systems for shutdown conditions as a backup to the normal residual heat removal system. This reduces the risk at shutdown through redundancy and diversity.

Post-72-hour connections are incorporated into the passive system design to allow for long-term accident management. These connections allow for the refill of the in-containment refueling water storage tank, or the reactor cavity, should such actions become necessary.

59.9.2.2 Nonsafety-Related Systems

AP1000 has nonsafety-related systems capable of mitigating accidents. These systems use redundant components, which are powered by offsite and onsite power supplies. AP1000 has certain design features in the nonsafety-related systems to reduce plant risk compared to current operating plants. During transient events, the startup feedwater system can act as a backup to the main feedwater system if the latter is unavailable due to the nature of the initiating event or fails during the transient. During loss of ac power events, startup feedwater pumps are powered by the diesel generators and can be used to remove decay heat since main feedwater is not available. The main feedwater and startup feedwater pumps are motor-driven, rather than steam-driven, for better reliability. Main feedwater controls are digital for better reliability. Thus, the main feedwater and startup feedwater system creates fewer transients and provides additional nonsafety-related means for decay heat removal for transients. This makes the plant response to transients very robust due to the existence of two nonsafety-related systems in addition to the passive safety-related means of removing decay heat.

The nonsafety-related normal residual heat removal system plays a role in decay heat removal in response to power and shutdown events. The normal residual heat removal system has additional isolation valves and is designed to withstand the reactor coolant system pressure to eliminate interfacing systems loss-of-coolant accident concerns that lead to containment bypass. The normal residual heat removal system provides reliable shutdown cooling, incorporating lessons learned from shutdown events. During mid-loop operations, operation procedures require both normal residual heat removal system pumps to be operable for risk reduction.

Component cooling water and service water systems have a very limited role in the plant risk profile because the passive safety-related systems do not require cooling, and the canned-motor reactor coolant pumps do not require seal cooling from the component cooling water.

The nonsafety-related ac power system (onsite and offsite) also has a very limited role in the plant risk profile since the plant safety-related systems do not depend on ac power. The loss of offsite power event is less important for the AP1000 than in current operating plants. The plant has full load rejection capability to minimize the number of reactor trips although this is not modeled in the PRA and no credit is taken for it. The onsite ac power has two nonsafety-related diesel generators. The diesel generator life is improved and the run failure rate is reduced by avoiding fast starts.

The compressed and instrument air system has low risk importance since the safety-related air-operated valves are fail safe if the air system fails. This causes the loss of air event to be less important than in current plant PRAs.

59.9.3 Instrumentation and Control Design

Three instrumentation and control systems are modeled in the AP1000 PRA: protection and safety monitoring system, plant control system, and diverse actuation system. Both the protection and safety monitoring system and plant control system are microprocessor-based. Four trains of redundancy are provided for the protection and safety monitoring system; 2-out-of-4 actuation logic in the protection and safety monitoring system reduces the potential for spurious trips due to testing and allows for better testing. Automatic testing for the protection and safety monitoring system, and diagnostic self-testing for the protection and safety monitoring system and the plant control system, provide higher reliability in these systems. Both the protection and safety monitoring system and the plant control system use fiber-optic cables (with fire separation) for data transmission. Unlike current plants, there is no cable spreading room, thus eliminating a potential fire hazard. Additional fault tolerance is built into the plant control system so that one failure does not prevent the operation of important functions.

Improvements in the plant control system and the protection and safety monitoring system are coupled with an improved control room and man-machine interfaces; these include improvements in the form and contents of the information provided to control room operators for decision making to limit commission errors. In addition, the remote shutdown workstation is designed to have functions similar to the control room.

The diverse actuation system provides a diverse automatic and manual backup function to the protection and safety monitoring system and reduces risk from anticipated transients without scram events. The diverse actuation system also compensates for common cause failures in the protection and safety monitoring system.

59.9.4 Plant Layout

The plant layout minimizes the consequences of fire and flooding by maximizing the separation of electrical and mechanical equipment areas in the non-radiologically controlled area of the auxiliary building. This separation is designed to minimize the potential for propagation of leaks from the piping areas and the mechanical equipment areas to the Class 1E electrical and Class IE instrumentation and control equipment rooms. The potential flooding sources and volumes in areas of the plant that contain safety-related electrical and I&C equipment are limited to minimize the consequences of internal flooding.

AP1000 is designed to provide better separation between divisions of safety-related equipment.

59.9.5 Containment Design

The containment pressure boundary is the final barrier to the release of fission products to the environment. The AP1000 containment has provisions that help to maintain containment integrity in the event of a severe accident.

59.9.5.1 Containment Isolation and Leakage

Failure of the containment isolation system prior to a severe accident will lead to a direct release pathway from the containment volume to the environment. AP1000 has approximately 55 percent fewer piping penetrations and a lower percentage of normally open penetrations compared to current generation plants. Normally open penetrations are closed by automatic valves, and diverse actuation is provided for valves on penetrations with significant leakage potential. All isolation valves have control room indication to inform the operator of the current valve position.

Similarly to containment isolation failure, leakage of closed containment isolation valves in excess of technical specifications may result in larger releases to the environment. Valves that historically have the greatest leakage problems have been eliminated, or their number significantly reduced in the design. Large purge valves have been replaced by smaller more reliable valves, and check valves have only been used in mild service where wear and service conditions would not be a challenge to successful operation.

Equipment and personnel hatches have the capability of being tested individually to ensure a leak-tight seal. Hatch seals can easily be verified.

Therefore, AP1000 provides significant protection against the failure to isolate the containment and against failure of isolation valves to fully close.

59.9.5.2 Containment Bypass

Historically, containment bypass, an accident in which the fission products are released directly to the environment from the reactor coolant system, is the leading contributor to risk in a nuclear power plant. Typically the containment bypass accident class consists of two types of accident sequences: interfacing systems loss-of-coolant accidents and steam generator tube ruptures.

An interfacing systems loss-of-coolant accident is the failure of valves that separate the high pressure reactor coolant system with a lower pressure interfacing system, which extends outside the containment pressure boundary. The failure of the valve causes the reactor coolant system to pressurize the interfacing system beyond its ultimate capacity and can result in a loss-of-coolant accident outside the containment. Reactor coolant is lost outside the containment, providing a pathway for the direct release of fission products to the environment. In AP1000, systems connected to the reactor coolant system are designed with higher design pressure, which reduces the likelihood of a pipe rupture in the event of the failure of the interfacing valves. This results in a very low interfacing systems loss-of-coolant-accident contribution to core damage to containment bypass.

Steam generator tube ruptures release coolant from the reactor coolant system to the secondary system. The AP1000 has multiple and diverse automatically actuated systems to reduce the reactor coolant system pressure and mitigate the steam generator tube rupture. The passive residual heat removal subsystem is actuated automatically on the S-signal and effectively reduces the reactor coolant system pressure to stop the break flow. If the passive residual heat removal does not stop the loss of coolant, the secondary relief valve can open to keep the secondary system pressure below the opening pressure of the steam generator safety valve. If the loss of reactor coolant continues, the RCS automatic depressurization system will actuate and depressurize the system. No operator actions are required to mitigate the accident, and the secondary system remains sealed against releases to the environment after the relief valve or its block valve are closed.

To create a containment bypass release pathway from a steam generator tube rupture, the accident scenario must include multiple system failures such that the steam generator tube rupture is not mitigated, and the secondary system pressure increases enough to open a safety valve. The safety valve must fail to reseat, thereby providing a containment bypass pathway for the loss of coolant and for the possible release of fission products to the environment.

Multiple, diverse systems act to mitigate steam generator tube rupture. Therefore, the likelihood of a steam generator tube rupture progressing to containment bypass has been significantly reduced in AP1000.

59.9.5.3 Passive Containment Cooling

The passive containment cooling system provides protection to the containment pressure boundary by removing the decay and chemical heat that slowly pressurize the containment. The heat is transferred to the environment through the steel pressure boundary. The heat transfer on the outside of the steel shell is enhanced by an annular flow path, which creates a convective air flow across the shell and by the evaporation of water that is directed onto the top of the containment in the event of an accident. The evaporative heat transfer prevents the containment from pressurizing above the design conditions during design basis accidents.

In some postulated multiple-failure accident scenarios, the water flow may fail. The heat removal is limited to convection heat transfer to the air flow and radiation to the annulus baffle. With no water film on the containment shell to provide evaporative cooling, the containment pressurizes above the design pressure to remove decay heat. Containment failure within 24 hours is highly unlikely.

59.9.5.4 High-Pressure Core Melt Scenarios

The automatic depressurization system and the passive residual heat removal heat exchanger provide reliable and diverse reactor coolant system depressurization, which significantly reduces the likelihood of high pressure core damage. High-pressure core damage sequences have the potential to fail steam generator tubes and create a containment bypass release, or to cause severe accident phenomena at the time of vessel failure which may threaten the containment pressure boundary. Reducing the reactor coolant system pressure during a severe accident significantly lowers the likelihood of phenomena that may induce large fission product releases early in the accident sequence.

59.9.5.5 In-Vessel Retention of Molten Core Debris

The AP1000 reactor vessel and containment configuration have features that enhance the design's ability to maintain molten core debris in the reactor vessel. The AP1000 automatic depressurization system provides reliable pressure reduction in the reactor coolant system to reduce the stresses on the vessel wall. The reactor vessel lower head has no vessel penetrations, thus eliminating penetration failure as a potential vessel failure mode. The containment configuration directs water to the reactor cavity and allows the in-containment refueling water storage tank water to be drained into the cavity to submerge the vessel to cool the external surface of the lower head. Cooling the vessel and reducing the stresses prevents the creep rupture failure of the vessel wall. The reactor vessel reflective insulation has been designed with provisions to allow water inside the insulation panel to cool the vessel surface, and with vents to allow steam to exit the insulation without failing the insulation support structures. The insulation is designed so that it promotes the cooling of the external surface of the vessel.

Preventing the relocation of molten core debris to the containment eliminates the occurrence of several severe accident phenomena, such as ex-vessel fuel-coolant interactions and core-concrete interaction, which may threaten the containment integrity. Through the prevention of core debris relocation to the containment, the AP1000 design significantly reduces the likelihood of containment failure.

59.9.5.6 Combustible Gases Generation and Burning

In severe accident sequences, high temperature metal oxidation, particularly zirconium, results in the rapid generation of hydrogen and possibly carbon monoxide. The first combustible gas release occurs in the accident sequence during core uncovery when the oxidation of the zircaloy cladding by passing steam generates hydrogen. A second release may occur if the vessel fails and ex-vessel debris degrades the concrete basemat. Steam and carbon dioxide are liberated from the concrete and are reduced to hydrogen and carbon monoxide as they pass through the molten metal in the debris. These gases are highly combustible and in high concentrations in the containment may lead to detonable mixtures.

The AP1000 uses a nonsafety-related hydrogen igniter system for severe releases of combustible gases. The igniters are powered from ac buses from either of the nonsafety-related diesel generators or from the non-Class 1E batteries. Multiple glow plugs are located in each compartment. The igniters burn the gases at the lower flammability limit. At this low concentration, the containment pressure increase from the burning is small and the likelihood of detonation is negligible. The igniters are spaced such that the distance between them will not allow the burn to transition from deflagration to detonation. The combustible gases are removed with no threat to the containment integrity.

There is little threat of the failure of the system power in the event that it is required to operate. The igniters are needed only in core damage accidents, and the AP1000 is designed to mitigate loss of power events without the sequence evolving into a severe accident. Loss of ac power is a small contributor to the core damage frequency.

The reliability of reactor coolant system depressurization reduces the threat to the containment from sudden releases of hydrogen from the reactor coolant system. Low pressure release of in-vessel hydrogen enhances the ability of the igniter system to maintain the containment atmosphere at the lower flammability limit.

During a severe accident, hydrogen that could be injected from the reactor coolant system into the containment through the spargers in the in-containment refueling water storage tank or into the core makeup tank room has the potential to produce a diffusion flame. A diffusion flame is produced when a combustible gas plume that is too rich to burn enters an oxygen-rich atmosphere and is ignited by a glow plug or a random ignition source. The plume is ignited into a standing flame which lasts as long as there is a fuel source. Via convection and radiation, the flame can heat the containment wall to high temperatures, increasing the likelihood of creep rupture failure of the containment pressure boundary. The AP1000 uses a defense-in-depth approach to release hydrogen in benign locations away from the containment shell and penetrations. Therefore, the potential for containment failure from the formation of a diffusion flame at the in-containment refueling water storage tank vents is considered to be very low.

There is little threat to the containment integrity from severe accident hydrogen releases, and hydrogen combustion events. The igniter system maintains the hydrogen concentration at the lower flammability limit.

59.9.5.7 Intermediate and Long-Term Containment Failure

The passive containment cooling system reduces the potential for decay heat pressurization of the containment. However, containment failure can also occur as a result of combustion. Due to the high likelihood of in-vessel retention of core debris, the potential for ex-vessel combustible gas generation from core-concrete interaction is very low. The frequency of containment failures due to hydrogen combustion events is very low given the high reliability of the hydrogen igniters.

59.9.5.8 Fission-Product Removal

AP1000 relies on the passive, natural removal of aerosol fission products from the containment atmosphere, primarily from gravitational settling, diffusiophoresis and thermophoresis. Natural removal is enhanced by the passive containment cooling system, which provides a large, cold surface area for condensation of steam. This increases the diffusiophoretic and thermophoretic removal processes. Accident offsite doses at the site boundary that could exist in the first 24 hours after a severe accident are either less than 25 rem, or for those releases that are greater than 25 rem, have a frequency of much less than 1E-06. Minimal credit is taken for deposition of fission products in the auxiliary building. The site boundary dose and large release frequency are much less than the established goals.

59.10 PRA Input to Design Certification Process

The AP1000 PRA was used in the design certification process to identify important safety insights and assumptions to support certification requirements, such as the reliability assurance program (RAP).

59.10.1 PRA Input to Reliability Assurance Program

The AP1000 RAP identifies those systems, structures, and components (SSC) that should be given priority in maintaining their reliability through surveillance, maintenance, and quality control actions during plant operation. The PRA importance and sensitivity analyses identify those systems and components important in plant risk in terms of either risk increase (for example, what happens to plant risk if a system or component, or a train is unavailable), or in terms of risk decrease (for example, what happens to plant risk if a component or a train is perfectly reliable/available). This ranking of components and systems in such a way provides an input for the reliability assurance program. For more information on the AP1000 reliability assurance program, refer to AP1000 DCD Section 17.4.

59.10.2 PRA Input to Tier 1 Information

AP1000 DCD Section 14.3 summarizes the design material contained in AP1000 that has been incorporated into the AP1000 DCD Tier 1 Information from the PRA.

59.10.3 PRA Input to MMI/Human Factors/Emergency Response Guidelines

The PRA models, including modeling of operator actions in response to severe accident sequences, follow the ERGs. The most risk-important of these actions is manual actuation of systems in the highly unlikely event of automatic actuation failure. These operator actions and the main human reliability analysis (HRA) model assumptions are reviewed by human factors engineers for insights that they may provide to the human system interface (HSI) and human factors areas. For more information on the AP1000 HSI, refer to AP1000 DCD Chapter 18.

In addition, the human reliability analysis models and operator actions modeled in the PRA were reviewed by the engineers writing the ERGs for consistency between the PRA models and the actual ERGs.

The PRA results and sensitivity studies show that the AP1000 design has no critical operator actions and few risk-important actions. A critical operator action is defined as that action when assumed to fail would result in a plant core damage frequency of greater than 1.0E-04 per year; there are no such operator actions in the AP1000 PRA.

59.10.4 Summary of PRA Based Insights

The use of the PRA in the design process is discussed in Section 59.2. A summary of the overall PRA results is provided in Sections 59.3 through 59.8. A discussion of the AP1000 plant features important to reducing risk is provided in Section 59.9. PRA-based insights are developed from this information and are summarized in Table 59-18.

59.10.5 Combined License Information

The Combined License applicant referencing the AP1000 certified design will review differences between the as-built plant and the design used as the basis for the AP1000 seismic margins analysis. Differences will be evaluated to determine if there is significant adverse effect on the seismic margins analysis results. Spacial interactions are addressed by COL information item 3.7-3. Details of the process will be developed by the Combined License applicant.

The Combined License applicant referencing the AP1000 certified design should compare the as-built SSC HCLPFs to those assumed in the AP1000 seismic margin evaluation. Deviations from the HCLPF values or assumptions in the seismic margin evaluation should be evaluated to determine if vulnerabilities have been introduced.

The Combined License applicant referencing the AP1000 certified design will review differences between the as-built plant and the design used as the basis for the AP1000 PRA and Table 59-18. If the effects of the differences are shown, by a screening analysis, to potentially result in a significant increase in core damage frequency or large release frequency, the PRA will be updated to reflect these differences.

The Combined License applicant referencing the AP1000 certified design will review differences between the as-built plant and the design used as the basis for the AP1000 internal fire and internal flood analysis. Differences will be evaluated to determine if there is significant adverse effect on the internal fire and internal flood analysis results.

The Combined License applicant referencing the AP1000 certified design will develop and implement severe accident management guidance using the suggested framework provided in WCAP-13914, "Framework for AP600 Severe Accident Management Guidance," (Reference 59-1).

The Combined License applicant referencing the AP1000 certified design will perform a thermal lag assessment of the as-built equipment required to mitigate severe accidents (hydrogen igniters and containment penetrations) to provide additional assurance that this equipment can perform its severe accident functions during environmental conditions resulting from hydrogen burns associated with severe accidents. This assessment is required only for equipment used for severe accident mitigation that has not been tested at severe accident conditions. The Combined License applicant will assess the ability of the as-built equipment to perform during severe accident hydrogen burns, using the Environment Enveloping method or the Test Based Thermal Analysis method discussed in EPRI NP-4354 (Reference 59-2).

59.11 References

59-1 "Framework for AP600 Severe Accident Management Guidance," WCAP-13914, Revision 3, January 1998.

59-2 "Large Scale Hydrogen Burn Equipment Experiments," EPRI-NP-4354, December 1985.

Table 59-1

CONTRIBUTION OF INITIATING EVENTS TO CORE DAMAGE

	Core Damage Contribution	Initiating Event Category	Percent Contribution	Initiating Event Frequency
1	9.50E-08	SAFETY INJECTION LINE BREAK INITIATING EVENT	39 4%	2.12E-04
2	4.50E-08	LARGE LOCA INITIATING EVENT	18.7%	5.00E-06
3	2.96E-08	SPURIOUS ADS INITIATING EVENT	12.3%	5.40E-05
4	1.81E-08	SMALL LOCA INITIATING EVENT	7.5%	5.00E-04
5	1.61E-08	MEDIUM LOCA INITIATING EVENT	6.7%	4.36E-04
6	1.00E-08	REACTOR VESSEL RUPTURE INITIATING EVENT	4.2%	1.00E-08
7	6.79E-09	STEAM GENERATOR TUBE RUPTURE INITIATING EVENT	2.8%	3.88E-03
8	3.68E-09	CMT LINE BREAK INITIATING EVENT	1.5%	9.31E-05
9	3.61E-09	ATWS PRECURSOR WITH NO MFW INITIATING EVENT	1.5%	4.81E-01(*)
10	3.08E-09	TRANSIENT WITH MFW INITIATING EVENT	1.3%	1.40E+00
11	1.71E-09	RCS LEAK INITIATING EVENT	0.7%	6 20E-03
12	1.66E-09	CORE POWER EXCURSION INITIATING EVENT	0.7%	4.50E-03
13	1.24E-09	LOSS OF CONDENSER INITIATING EVENT	0.5%	1.12E-01
14	9.58E-10	LOSS OF OFFSITE POWER INITIATING EVENT	0.4%	1.20E-01
15	8.70E-10	LOSS OF MAIN FEEDWATER INITIATING EVENT	0.4%	3.35E-01
16	7.12E-10	ATWS PRECURSOR WITH MFW AVAILABLE INITIATING EVENT	0.3%	1.17E+00(*)
17	6.72E-10	LOSS OF COMPRESSED AIR INITIATING EVENT	0.3%	3.48E-02
18	6 06E-10	MAIN STEAM LINE STUCK-OPEN SV INITIATING EVENT	0.3%	2.39E-3
19	5.02E-10	PASSIVE RHR TUBE RUPTURE INITIATING EVENT	0.2%	1.34E-04
20	4.53E-10	LOSS OF MFW TO ONE SG INITIATING EVENT	0.2%	1.92E-01
21	3.23E-10	LOSS OF CCW/SW INITIATING EVENT	0.1%	1.44E-01
22	1.31E-10	MAIN STEAM LINE BREAK UPSTREAM OF MSIV INITIATING EVENT	0.1%	3.72E-04
23	1.11E-10	ATWS PRECURSOR WITH SI SIGNAL INITIATING EVENT	0 1%	1.48E-02(*)
24	5.00E-11	INTERFACING SYSTEMS LOCA INITIATING EVENT	0 0%	5.00E-11
25	3.52E-11	LOSS OF RCS FLOW INITIATING EVENT	0 0%	1.80E-02
26	9.15E-12	MAIN STEAM LINE BREAK DOWNSTREAM OF MSIV INITIATING EVENT	0 0%	5.96E-04
	2.41E-07	Totals	100.0%	2.38(*)

(*) Note that the ATWS precursor frequencies are not included in the total initiating event frequency, since they are already accounted for in the other categories.

Table 59-2

CONDITIONAL CORE DAMAGE PROBABILITY OF INITIATING EVENTS

	Core Damage Contribution	Initiating Event Category	Initiating Event Frequency	Conditional CD Prob.
6	1.00E-08	REACTOR VESSEL RUPTURE INITIATING EVENT	1.00E-08	1 00E+00
24	5.00E-11	INTERFACING SYSTEMS LOCA INITIATING EVENT	5.00E-11	1 00E+00
2	4.50E-08	LARGE LOCA INITIATING EVENT	5 00E-06	8.99E-03
3	2 96E-08	SPURIOUS ADS INITIATING EVENT	5.40E-05	5 48E-04
1	9.50E-08	SAFETY INJECTION LINE BREAK INITIATING EVENT	2.12E-04	4 48E-04
8	3 68E-09	CMT LINE BREAK INITIATING EVENT	9.31E-05	3.95E-05
5	1.61E-08	MEDIUM LOCA INITIATING EVENT	4.36E-04	3.70E-05
4	1.81E-08	SMALL LOCA INITIATING EVENT	5.00E-04	3.62E-05
19	5.02E-10	PASSIVE RHR TUBE RUPTURE INITIATING EVENT	1 34E-04	3.74E-06
7	6.79E-09	STEAM GENERATOR TUBE RUPTURE INITIATING EVENT	3.88E-03	1.75E-06
18	6 06E-10	MAIN STEAM LINE STUCK-OPEN SV INITIATING EVENT	2.39E-03	2.54E-07
12	1.66E-09	CORE POWER EXCURSION INITIATING EVENT	4.50E-03	3.69E-07
22	1.31E-10	MAIN STEAM LINE BREAK UPSTREAM OF MSIV INITIATING EVENT	3.72E-04	3 51E-07
11	1.71E-09	RCS LEAK INITIATING EVENT	6 20E-03	2.75E-07
17	6.72E-10	LOSS OF COMPRESSED AIR INITIATING EVENT	3.48E-02	1.93E-08
26	9.15E-12	MAIN STEAM LINE BREAK DOWNSTREAM OF MSIV INITIATING EVENT	5.96E-04	1.54E-08
13	1.24E-09	LOSS OF CONDENSER INITIATING EVENT	1.12E-01	1.11E-08
14	9.58E-10	LOSS OF OFFSITE POWER INITIATING EVENT	1.20E-01	7.98E-09
9	3 61E-09	ATWS PRECURSOR WITH NO MFW INITIATING EVENT	4.81E-01	7.49E-09
23	1.11E-10	ATWS PRECURSOR WITH SI SIGNAL INITIATING EVENT	1.48E-02	7.48E-09
15	8 70E-10	LOSS OF MAIN FEEDWATER INITIATING EVENT	3.35E-01	2.60E-09
20	4 53E-10	LOSS OF MFW TO ONE SG INITIATING EVENT	1.92E-01	2.36E-09
21	3 23E-10	LOSS OF CCW/SW INITIATING EVENT	1.44E-01	2.24E-09
10	3.08E-09	TRANSIENT WITH MFW INITIATING EVENT	1.40E+00	2.20E-09
25	3.52E-11	LOSS OF RSC FLOW INITIATING EVENT	1 80E-02	1.96E-09
16	7.12E-10	ATWS PRECURSOR WITH MFW AVAILABLE INITIATING EVENT	1.17E+00	6 09E-10
	2 41E-07	Totals	2.38E+00	

Table 59-3 (Sheet 1 of 4)

INTERNAL INITIATING EVENTS AT POWER DOMINANT CORE DAMAGE SEQUENCES

	Sequence Frequency	Percent Contrib	Cumulative % Contrib	Sequence Identifier	Sequence Description
1	6.88E-08	28.52	28.52	2esil-07	SAFETY INJECTION LINE BREAK INITIATING EVENT OCCURS RCPS TRIP AND CMT INJECTION IS SUCCESSFUL – 1 OF 2 CMT TRAINS SUCCESS OF FULL ADS DEPRESSURIZATION FAILURE OF ONE OF ONE IRWST INJECTION LINE
2	4.26E-08	17.66	46.18	2rllo-09	LARGE LOCA INITIATING EVENT OCCURS ANY ONE OF TWO ACCUMULATOR TRAINS FAIL
3	2.13E-08	8.82	55.00	3dsad-08	SPURIOUS ADS INITIATING EVENT OCCURS SUCCESS OF 1/2 OR 2/2 ACCUMULATORS FAILURE OF ADS OR CMT
4	1.98E-08	8.23	63.23	3dsil-08	SAFETY INJECTION LINE BREAK INITIATING EVENT OCCURS RCPS TRIP AND CMT INJECTION IS SUCCESSFUL – 1 OF 2 CMT TRAINS FAILURE OF FULL ADS DEPRESSURIZATION
5	1.00E-08	4.15	67.38	3crvr-02	REACTOR VESSEL RUPTURE INITIATING EVENT OCCURS
6	8.44E-09	3.5	70.88	2lslo-05	SMALL LOCA INITIATING EVENT OCCURS SUCCESS OF CMT & RCP TRIP SUCCESS OF PASSIVE RHR SYSTEM SUCCESS OF FULL ADS DEPRESSURIZATION FAILURE OF NORMAL RHR IN INJECTION MODE SUCCESS OF TWO OF TWO IRWST INJECTION LINES SUCCESS OF CIS & PRE-EXISTING CONTAINMENT OPENING FAILURE OF RECIRCULATION

Table 59-3 (Sheet 2 of 4)

INTERNAL INITIATING EVENTS AT POWER DOMINANT CORE DAMAGE SEQUENCES

	Sequence Frequency	Percent Contrib	Cumulative % Contrib	Sequence Identifier	Sequence Description
7	7.35E-09	3.05	73.93	2lmlo-05	MEDIUM LOCA INITIATING EVENT OCCURS SUCCESS OF CMT & RCP TRIP SUCCESS OF FULL ADS DEPRESSURIZATION FAILURE OF NORMAL RHR IN INJECTION MODE SUCCESS OF TWO OF TWO IRWST INJECTION LINES SUCCESS OF CIS & PRE-EXISTING CONTAINMENT OPENING FAILURE OF RECIRCULATION
8	5.11E-09	2.12	76.05	3dslo-12	SMALL LOCA INITIATING EVENT OCCURS SUCCESS OF CMT & RCP TRIP SUCCESS OF PASSIVE RHR SYSTEM FAILURE OF FULL ADS DEPRESSURIZATION SUCCESS OF PARTIAL ADS DEPRESSURIZATION FAILURE OF NORMAL RHR IN INJECTION MODE
9	4.46E-09	1.85	77.90	3dmlo-12	MEDIUM LOCA INITIATING EVENT OCCURS SUCCESS OF CMT & RCP TRIP FAILURE OF FULL ADS DEPRESSURIZATION SUCCESS OF PARTIAL ADS DEPRESSURIZATION FAILURE OF NORMAL RHR IN INJECTION MODE
10	3.72E-09	1.54	79.44	2rsad-09	SPURIOUS ADS INITIATING EVENT OCCURS FAILURE OF 2/2 ACCUMULATORS
11	3.67E-09	1.52	80.96	2esad-07	SPURIOUS ADS INITIATING EVENT OCCURS SUCCESS OF 1/2 OR 2/2 ACCUMULATORS SUCCESS OF ADS & CMT FAILURE OF IRW OR CMT

Table 59-3 (Sheet 3 of 4)

INTERNAL INITIATING EVENTS AT POWER DOMINANT CORE DAMAGE SEQUENCES

	Sequence Frequency	Percent Contrib	Cumulative % Contrib	Sequence Identifier	Sequence Description
12	3.57E-09	1.48	82.44	2lsil-03	SAFETY INJECTION LINE BREAK INITIATING EVENT OCCURS RCPS TRIP AND CMT INJECTION IS SUCCESSFUL – 1 OF 2 CMT TRAINS SUCCESS OF FULL ADS DEPRESSURIZATION IRWST INJECTION IS SUCCESSFUL – 1 OF 1 TRAINS SUCCESS OF CIS & PRE-EXISTING CONTAINMENT OPENING FAILURE OF RECIRCULATION
13	3.55E-09	1.47	83.91	6esgt-41	SGTR EVENT SEQUENCE CONTINUES FAILURE OF CMT OR RCP TRIP SUCCESS OF PASSIVE RHR SYSTEM FAILURE OF FULL ADS DEPRESSURIZATION FAILURE OF PARTIAL ADS DEPRESSURIZATION
14	3.31E-09	1.37	85.28	3aatw-23	ATWS PRECURSOR WITH NO MFW EVENT SEQUENCE CONTINUES SUCCESS OF SFW OR PRHR SYSTEM SUCCESS OF MANUAL REACTOR TRIP FAILURE OF MANUAL BORATION BY CVS FAILURE OF CMT OR RCP TRIP
15	3.30E-09	1.37	86.65	2eslo-09	SMALL LOCA INITIATING EVENT OCCURS SUCCESS OF CMT & RCP TRIP SUCCESS OF PASSIVE RHR SYSTEM SUCCESS OF FULL ADS DEPRESSURIZATION FAILURE OF NORMAL RHR IN INJECTION MODE FAILURE OF TWO OF TWO IRWST INJECTION LINES

Table 59-3 (Sheet 4 of 4)

INTERNAL INITIATING EVENTS AT POWER DOMINANT CORE DAMAGE SEQUENCES

	Sequence Frequency	Percent Contrib	Cumulative % Contrib	Sequence Identifier	Sequence Description
16	2.88E-09	1.19	87.84	2cmlo-09	MEDIUM LOCA INITIATING EVENT OCCURS SUCCESS OF CMT & RCP TRIP SUCCESS OF FULL ADS DEPRESSURIZATION FAILURE OF NORMAL RHR IN INJECTION MODE FAILURE OF TWO OF TWO IRWST INJECTION LINES
17	2.19E-09	0.91	88.75	6esgt-13	SGTR EVENT SEQUENCE CONTINUES SUCCESS OF CMT & RCP TRIP SUCCESS OF PASSIVE RHR SYSTEM FAILURE OF FULL ADS DEPRESSURIZATION FAILURE OF PARTIAL ADS DEPRESSURIZATION
18	1.97E-09	0.82	89.57	3dllo-08	LARGE LOCA INITIATING EVENT OCCURS ACCUMULATOR INJECTION IS SUCCESSFUL – 2 OF 2 TRAINS FAILURE OF ADS OR CMT
19	1.57E-09	0.65	90.22	2lcmt-05	CMT LINE BREAK INITIATING EVENT OCCURS RCPS TRIP AND CMT INJECTION IS SUCCESSFUL – 1 OF 2 CMT TRAINS SUCCESS OF FULL ADS DEPRESSURIZATION FAILURE OF NORMAL RHR IN INJECTION MODE SUCCESS OF TWO OF TWO IRWST INJECTION LINES SUCCESS OF CIS & PRE-EXISTING CONTAINMENT OPENING FAILURE OF RECIRCULATION

Table 59-4 (Sheet 1 of 3)

SEQUENCE 1 – SAFETY INJECTION LINE BREAK DOMINANT CUTSETS (SI-LB-07)

NUMBER	CUTSET PROB	PERCENT	BASIC EVENT NAME		
1	5.09E-08	74.04	SAFETY INJECTION LINE BREAK INITIATING EVENT OCCURS	2.12E-04	IEV-SI-LB
			IWRST DISCHARGE LINE "A" STRAINER PLUGGED	2.40E-04	IWA-PLUG
2	6.36E-09	9.25	SAFETY INJECTION LINE BREAK INITIATING EVENT OCCURS	2.12E-04	IEV-SI-LB
			CCF OF 4 GRAVITY INJECTION CVs	3.00E-05	IWX-CV-AO
3	5.51E-09	8.01	SAFETY INJECTION LINE BREAK INITIATING EVENT OCCURS	2.12E-04	IEV-SI-LB
			CCF OF 4 GRAVITY INJECTION & 2 RECIRCULATION SQUIB VALVES	2.60E-05	IWX-EV-SA
4	1.23E-09	1.79	SAFETY INJECTION LINE BREAK INITIATING EVENT OCCURS	2.12E-04	IEV-SI-LB
			CCF OF 2 GRAVITY INJECTION SQUIB VALVES IN 1/1 LINES TO OPEN	5.80E-06	IWX-EV1-SA
5	6.49E-10	.94	SAFETY INJECTION LINE BREAK INITIATING EVENT OCCURS	2.12E-04	IEV-SI-LB
			CHECK VALVE 122A FAILS TO OPEN	1.75E-03	IWACV122AO
			CHECK VALVE 124A FAILS TO OPEN	1.75E-03	IWACV124AO
6	5.42E-10	.79	SAFETY INJECTION LINE BREAK INITIATING EVENT OCCURS	2.12E-04	IEV-SI-LB
			CHECK VALVE 122A FAILS TO OPEN	1.75E-03	IWACV122AO
			HARDWARE FAILURE OF VALVE 125A	1.46E-03	IRWMOD06
7	5.42E-10	.79	SAFETY INJECTION LINE BREAK INITIATING EVENT OCCURS	2.12E-04	IEV-SI-LB
			HARDWARE FAILURE OF VALVE 123A	1.46E-03	IRWMOD05
			CHECK VALVE 124A FAILS TO OPEN	1.75E-03	IWACV124AO
8	4.52E-10	.66	SAFETY INJECTION LINE BREAK INITIATING EVENT OCCURS	2.12E-04	IEV-SI-LB
			HARDWARE FAILURE OF VALVE 123A	1.46E-03	IRWMOD05
			HARDWARE FAILURE OF VALVE 125A	1.46E-03	IRWMOD06
9	3.25E-10	.47	SAFETY INJECTION LINE BREAK INITIATING EVENT OCCURS	2.12E-04	IEV-SI-LB
			CHECK VALVE 122A FAILS TO OPEN	1.75E-03	IWACV122AO
			RELAY FAILS TO OPERATE	8.76E-04	IWDRS125AFA
10	3.25E-10	.47	SAFETY INJECTION LINE BREAK INITIATING EVENT OCCURS	2.12E-04	IEV-SI-LB
			CHECK VALVE 124A FAILS TO OPEN	1.75E-03	IWACV124AO
			RELAY FAILS TO OPERATE	8.76E-04	IWBRS123AFA
11	2.71E-10	.39	SAFETY INJECTION LINE BREAK INITIATING EVENT OCCURS	2.12E-04	IEV-SI-LB
			HARDWARE FAILURE OF VALVE 123A	1.46E-03	IRWMOD05
			RELAY FAILS TO OPERATE	8.76E-04	IWDRS125AFA

Table 59-4 (Sheet 2 of 3)

SEQUENCE 1 – SAFETY INJECTION LINE BREAK DOMINANT CUTSETS (SI-LB-07)

NUMBER	CUTSET PROB	PERCENT	BASIC EVENT NAME		
12	2.71E-10	.39	SAFETY INJECTION LINE BREAK INITIATING EVENT OCCURS	2.12E-04	IEV-SI-LB
			HARDWARE FAILURE OF VALVE 125A	1.46E-03	IRWMOD06
			RELAY FAILS TO OPERATE	8.76E-04	IWBRS123AFA
13	1.63E-10	.24	SAFETY INJECTION LINE BREAK INITIATING EVENT OCCURS	2.12E-04	IEV-SI-LB
			RELAY FAILS TO OPERATE	8.76E-04	IWBRS123AFA
			RELAY FAILS TO OPERATE	8.76E-04	IWDRS125AFA
14	1.14E-10	.17	SAFETY INJECTION LINE BREAK INITIATING EVENT OCCURS	2.12E-04	IEV-SI-LB
			CCF OF GRAVITY INJECTION CVs IN 1/1 LINES TO OPEN	5.40E-07	IWX-CV1-AO
15	1.11E-10	.16	SAFETY INJECTION LINE BREAK INITIATING EVENT OCCURS	2.12E-04	IEV-SI-LB
			CHECK VALVE 122A FAILS TO OPEN	1.75E-03	IWACV122AO
			BUS UNAVAILABLE DUE TO TEST OR CORRECTIVE MAINTENANCE	3.00E-04	IDDBSDS1TM
16	1.11E-10	.16	SAFETY INJECTION LINE BREAK INITIATING EVENT OCCURS	2.12E-04	IEV-SI-LB
			CHECK VALVE 122A FAILS TO OPEN	1.75E-03	IWACV122AO
			BUS UNAVAILABLE DUE TO TEST OR CORRECTIVE MAINTENANCE	3.00E-04	IDDBSDD1TM
17	1.11E-10	.16	SAFETY INJECTION LINE BREAK INITIATING EVENT OCCURS	2.12E-04	IEV-SI-LB
			CHECK VALVE 124A FAILS TO OPEN	1.75E-03	IWACV124AO
			BUS UNAVAILABLE DUE TO TEST OR CORRECTIVE MAINTENANCE	3.00E-04	IDBBSDS1TM
18	1.11E-10	.16	SAFETY INJECTION LINE BREAK INITIATING EVENT OCCURS	2.12E-04	IEV-SI-LB
			CHECK VALVE 124A FAILS TO OPEN	1.75E-03	IWACV124AO
			BUS UNAVAILABLE DUE TO TEST OR CORRECTIVE MAINTENANCE	3.00E-04	IDBBSDD1TM
19	9.29E-11	.14	SAFETY INJECTION LINE BREAK INITIATING EVENT OCCURS	2.12E-04	IEV-SI-LB
			HARDWARE FAILURE OF VALVE 123A	1.46E-03	IRWMOD05
			BUS UNAVAILABLE DUE TO TEST OR CORRECTIVE MAINTENANCE	3.00E-04	IDDBSDS1TM
20	9.29E-11	.14	SAFETY INJECTION LINE BREAK INITIATING EVENT OCCURS	2.12E-04	IEV-SI-LB
			HARDWARE FAILURE OF VALVE 123A	1.46E-03	IRWMOD05
			BUS UNAVAILABLE DUE TO TEST OR CORRECTIVE MAINTENANCE	3.00E-04	IDDBSDD1TM
21	9.29E-11	.14	SAFETY INJECTION LINE BREAK INITIATING EVENT OCCURS	2.12E-04	IEV-SI-LB
			HARDWARE FAILURE OF VALVE 125A	1.46E-03	IRWMOD06
			BUS UNAVAILABLE DUE TO TEST OR CORRECTIVE MAINTENANCE	3.00E-04	IDBBSDS1TM

Table 59-4 (Sheet 3 of 3)

SEQUENCE 1 – SAFETY INJECTION LINE BREAK DOMINANT CUTSETS (SI-LB-07)

NUMBER	CUTSET PROB	PERCENT	BASIC EVENT NAME		
22	9.29E-11	.14	SAFETY INJECTION LINE BREAK INITIATING EVENT OCCURS	2.12E-04	IEV-SI-LB
			HARDWARE FAILURE OF VALVE 125A	1.46E-03	IRWMOD06
			BUS UNAVAILABLE DUE TO TEST OR CORRECTIVE MAINTENANCE	3.00E-04	IDBBSDD1TM
23	5.57E-11	.08	SAFETY INJECTION LINE BREAK INITIATING EVENT OCCURS	2.12E-04	IEV-SI-LB
			RELAY FAILS TO OPERATE	8.76E-04	IWDRS125AFA
			BUS UNAVAILABLE DUE TO TEST OR CORRECTIVE MAINTENANCE	3.00E-04	IDBBSDS1TM
24	5.57E-11	.08	SAFETY INJECTION LINE BREAK INITIATING EVENT OCCURS	2.12E-04	IEV-SI-LB
			RELAY FAILS TO OPERATE	8.76E-04	IWDRS125AFA
			BUS UNAVAILABLE DUE TO TEST OR CORRECTIVE MAINTENANCE	3.00E-04	IDBBSDD1TM
25	5.57E-11	.08	SAFETY INJECTION LINE BREAK INITIATING EVENT OCCURS	2.12E-04	IEV-SI-LB
			RELAY FAILS TO OPERATE	8.76E-04	IWBRS123AFA
			BUS UNAVAILABLE DUE TO TEST OR CORRECTIVE MAINTENANCE	3.00E-04	IDDBSDS1TM

Table 59-5

SEQUENCE 2 – LARGE LOCA DOMINANT CUTSETS (LLOCA-09)

NUMBER	CUTSET PROB	PERCENT	BASIC EVENT NAME		
1	8.75E-09	20.55	LARGE LOCA INITIATING EVENT OCCURS	5.00E-06	IEV-LLOCA
			CHECK VALVE 029A FAILS TO OPEN	1.75E-03	ACACV029GO
2	8.75E-09	20.55	LARGE LOCA INITIATING EVENT OCCURS	5.00E-06	IEV-LLOCA
			CHECK VALVE 028A FAILS TO OPEN	1.75E-03	ACACV028GO
3	8.75E-09	20.55	LARGE LOCA INITIATING EVENT OCCURS	5.00E-06	IEV-LLOCA
			CHECK VALVE 029B FAILS TO OPEN	1.75E-03	ACBCV029GO
4	8.75E-09	20.55	LARGE LOCA INITIATING EVENT OCCURS	5.00E-06	IEV-LLOCA
			CHECK VALVE 028B FAILS TO OPEN	1.75E-03	ACBCV028GO
5	3.64E-09	8.55	LARGE LOCA INITIATING EVENT OCCURS	5.00E-06	IEV-LLOCA
			FLOW TUNING ORIFICE PLUGS	7.27E-04	ACAOR001SP
6	3.64E-09	8.55	LARGE LOCA INITIATING EVENT OCCURS	5.00E-06	IEV-LLOCA
			FLOW TUNING ORIFICE PLUGS	7.27E-04	ACBOR001SP
7	2.55E-10	.60	LARGE LOCA INITIATING EVENT OCCURS	5.00E-06	IEV-LLOCA
			COMMON CAUSE FAILURE OF 2 ACCUMULATOR CHECK VALVES	5.10E-05	ACX-CV-GO
8	1.20E-11	.03	LARGE LOCA INITIATING EVENT OCCURS	5.00E-06	IEV-LLOCA
			ACCUMULATOR TANK A (T001A) RUPTURES	2.40E-06	ACATK001AF
9	1.20E-11	.03	LARGE LOCA INITIATING EVENT OCCURS	5.00E-06	IEV-LLOCA
			ACCUMULATOR TANK B (T001B) RUPTURES	2.40E-06	ACBTK001AF
10	3.60E-12	.01	LARGE LOCA INITIATING EVENT OCCURS	5 00E-06	IEV-LLOCA
			FLOW TUNING ORIFICE RUPTURE	7.20E-07	ACAOR001EB
11	3.60E-12	.01	LARGE LOCA INITIATING EVENT OCCURS	5.00E-06	IEV-LLOCA
			FLOW TUNING ORIFICE RUPTURE	7.20E-07	ACBOR001EB
12	6.00E-13	.00	LARGE LOCA INITIATING EVENT OCCURS	5.00E-06	IEV-LLOCA
			COMMON CAUSE FAILURE OF ACCUMULATOR TANKS	1.20E-07	ACX-TK-AF

Table 59-6 (Sheet 1 of 3)

SEQUENCE 3 – SPURIOUS ADS ACTUATION DOMINANT CUTSETS (SPADS-08)

NUMBER	CUTSET PROB	PERCENT	BASIC EVENT NAME		
1	5.56E-09	26.14	SPURIOUS ADS INITIATING EVENT OCCURS	5.40E-05	IEV-SPADS
			CCF OF ESF INPUT LOGIC (HARDWARE)	1.03E-04	CCX-INPUT-LOGIC
2	3.35E-09	15.75	SPURIOUS ADS INITIATING EVENT OCCURS	5.40E-05	IEV-SPADS
			COMMON CAUSE FAILURE OF 4 AOVS TO OPEN	6.20E-05	CCX-AV-LA
3	3.19E-09	15.00	SPURIOUS ADS INITIATING EVENT OCCURS	5.40E-05	IEV-SPADS
			CCF OF 2 SQUIB VALVES TO OPERATE	5.90E-05	ADX-EV-SA2
4	2.75E-09	12.93	SPURIOUS ADS INITIATING EVENT OCCURS	5.40E-05	IEV-SPADS
			COMMON CAUSE FAILURE OF 4 CHECK VALVES TO OPEN	5.10E-05	CMX-CV-GO
5	2.07E-09	9.73	SPURIOUS ADS INITIATING EVENT OCCURS	5.40E-05	IEV-SPADS
			CCF OF RTD LEVEL TRANSMITTERS	3.84E-05	CMX-VS-FA
6	1.62E-09	7.62	SPURIOUS ADS INITIATING EVENT OCCURS	5.40E-05	IEV-SPADS
			DUE TO CCF OF 4TH STAGE ADS SQUIB VALVES TO OPERATE	3.00E-05	ADX-EV-SA
7	5.94E-10	2.79	SPURIOUS ADS INITIATING EVENT OCCURS	5.40E-05	IEV-SPADS
			CCF OF ESF INPUT LOGIC SOFTWARE	1.10E-05	CCX-IN-LOGIC-SW
8	5.94E-10	2.79	SPURIOUS ADS INITIATING EVENT OCCURS	5.40E-05	IEV-SPADS
			CCF OF PMS ESF ACTUATION LOGIC SOFTWARE	1.10E-05	CCX-PMXMOD2-SW
9	5.94E-10	2.79	SPURIOUS ADS INITIATING EVENT OCCURS	5.40E-05	IEV-SPADS
			CCF OF PMS ESF OUTPUT LOGIC SOFTWARE	1.10E-05	CCX-PMXMOD1-SW
10	4.65E-10	2.19	SPURIOUS ADS INITIATING EVENT OCCURS	5.40E-05	IEV-SPADS
			CCF OF EPO BOARDS IN PMS	8.62E-06	CCX-EP-SAM
11	6.48E-11	.30	SPURIOUS ADS INITIATING EVENT OCCURS	5.40E-05	IEV-SPADS
			SOFTWARE CCF OF ALL CARDS	1.20E-06	CCX-SFTW
12	2.85E-11	.13	SPURIOUS ADS INITIATING EVENT OCCURS	5.40E-05	IEV-SPADS
			FLOW TUNING ORIFICE PLUGS	7.27E-04	CMA-PLUG
			FLOW TUNING ORIFICE PLUGS	7.27E-04	CMB-PLUG
13	1.82E-11	.09	SPURIOUS ADS INITIATING EVENT OCCURS	5.40E-05	IEV-SPADS
			HARDWARE FAILURE OF ST. #4 LINE 3	5.80E-04	AD4MOD09
			HARDWARE FAILURE OF ST. #4 LINE 4	5.80E-04	AD4MOD10

Table 59-6 (Sheet 2 of 3)

SEQUENCE 3 – SPURIOUS ADS ACTUATION DOMINANT CUTSETS (SPADS-08)

NUMBER	CUTSET PROB	PERCENT	BASIC EVENT NAME		
14	1.82E-11	.09	SPURIOUS ADS INITIATING EVENT OCCURS	5.40E-05	IEV-SPADS
			HARDWARE FAILURE OF ST. #4 LINE 2	5.80E-04	AD4MOD08
			HARDWARE FAILURE OF ST. #4 LINE 4	5.80E-04	AD4MOD10
15	1.82E-11	.09	SPURIOUS ADS INITIATING EVENT OCCURS	5.40E-05	IEV-SPADS
			HARDWARE FAILURE OF ST. #4 LINE 2	5.80E-04	AD4MOD08
			HARDWARE FAILURE OF ST. #4 LINE 3	5.80E-04	AD4MOD09
16	1.82E-11	.09	SPURIOUS ADS INITIATING EVENT OCCURS	5.40E-05	IEV-SPADS
			HARDWARE FAILURE OF ST. #4 LINE 1	5.80E-04	AD4MOD07
			HARDWARE FAILURE OF ST. #4 LINE 4	5.80E-04	AD4MOD10
17	1.82E-11	.09	SPURIOUS ADS INITIATING EVENT OCCURS	5.40E-05	IEV-SPADS
			HARDWARE FAILURE OF ST. #4 LINE 1	5.80E-04	AD4MOD07
			HARDWARE FAILURE OF ST. #4 LINE 3	5.80E-04	AD4MOD09
18	1.82E-11	.09	SPURIOUS ADS INITIATING EVENT OCCURS	5.40E-05	IEV-SPADS
			HARDWARE FAILURE OF ST. #4 LINE 1	5.80E-04	AD4MOD07
			HARDWARE FAILURE OF ST. #4 LINE 2	5.80E-04	AD4MOD08
19	6.85E-12	.03	SPURIOUS ADS INITIATING EVENT OCCURS	5.40E-05	IEV-SPADS
			COMMON CAUSE FAILURE OF THE BATTERIES IDSA-DB-1A/1B	4.70E-05	CCX-BY-PN
			UNAVAILABILITY OF BUS ECS ES 2 DUE TO UNSCHEDULED MAINTENANCE	2.70E-03	EC2BS002TM
20	6.85E-12	.03	SPURIOUS ADS INITIATING EVENT OCCURS	5.40E-05	IEV-SPADS
			COMMON CAUSE FAILURE OF THE BATTERIES IDSA-DB-1A/1B	4.70E-05	CCX-BY-PN
			BUS UNAVAILABLE DUE TO UNSCHEDULED MAINTENANCE	2.70E-03	EC2BS022TM
21	6.85E-12	.03	SPURIOUS ADS INITIATING EVENT OCCURS	5.40E-05	IEV-SPADS
			COMMON CAUSE FAILURE OF THE BATTERIES IDSA-DB-1A/1B	4.70E-05	CCX-BY-PN
			BUS UNAVAILABLE DUE TO UNSCHEDULED MAINTENANCE	2.70E-03	EC2BS221TM
22	6.85E-12	.03	SPURIOUS ADS INITIATING EVENT OCCURS	5.40E-05	IEV-SPADS
			COMMON CAUSE FAILURE OF THE BATTERIES IDSA-DB-1A/1B	4.70E-05	CCX-BY-PN
			UNAVAILABILITY OF BUS ECS ES 1 DUE TO UNSCHEDULED MAINTENANCE	2.70E-03	EC1BS001TM
23	6.85E-12	.03	SPURIOUS ADS INITIATING EVENT OCCURS	5.40E-05	IEV-SPADS
			COMMON CAUSE FAILURE OF THE BATTERIES IDSA-DB-1A/1B	4.70E-05	CCX-BY-PN
			BUS UNAVAILABLE DUE TO UNSCHEDULED MAINTENANCE	2.70E-03	EC1BS012TM

Table 59-6 (Sheet 3 of 3)

SEQUENCE 3 – SPURIOUS ADS ACTUATION DOMINANT CUTSETS (SPADS-08)

NUMBER	CUTSET PROB	PERCENT	BASIC EVENT NAME		
24	6.85E-12	.03	SPURIOUS ADS INITIATING EVENT OCCURS	5.40E-05	IEV-SPADS
			COMMON CAUSE FAILURE OF THE BATTERIES IDSA-DB-1A/1B	4.70E-05	CCX-BY-PN
			BUS UNAVAILABLE DUE TO UNSCHEDULED MAINTENANCE	2.70E-03	EC1BS121TM
25	6.83E-12	.03	SPURIOUS ADS INITIATING EVENT OCCURS	5.40E-05	IEV-SPADS
			PMBMOD32	5.02E-03	PMBMOD32
			PMCMOD33	5.02E-03	PMCMOD33
			PMDMOD34	5.02E-03	PMDMOD34

Table 59-7 (Sheet 1 of 3)

SEQUENCE 4 – SAFETY INJECTION LINE BREAK DOMINANT CUTSETS (SI-LB-08)

NUMBER	CUTSET PROB	PERCENT	BASIC EVENT NAME		
1	1.25E-08	63.00	SAFETY INJECTION LINE BREAK INITIATING EVENT OCCURS	2.12E-04	IEV-SI-LB
			CCF OF 2 SQUIB VALVES TO OPERATE	5.90E-05	ADX-EV-SA2
2	6.36E-09	32.06	SAFETY INJECTION LINE BREAK INITIATING EVENT OCCURS	2.12E-04	IEV-SI-LB
			DUE TO CCF OF 4TH STAGE ADS SQUIB VALVES TO OPERATE	3.00E-05	ADX-EV-SA
3	7.13E-11	.36	SAFETY INJECTION LINE BREAK INITIATING EVENT OCCURS	2.12E-04	IEV-SI-LB
			HARDWARE FAILURE OF ST. #4 LINE 3	5.80E-04	AD4MOD09
			HARDWARE FAILURE OF ST. #4 LINE 4	5.80E-04	AD4MOD10
4	7.13E-11	.36	SAFETY INJECTION LINE BREAK INITIATING EVENT OCCURS	2.12E-04	IEV-SI-LB
			HARDWARE FAILURE OF ST. #4 LINE 2	5.80E-04	AD4MOD08
			HARDWARE FAILURE OF ST. #4 LINE 4	5.80E-04	AD4MOD10
5	7.13E-11	.36	SAFETY INJECTION LINE BREAK INITIATING EVENT OCCURS	2.12E-04	IEV-SI-LB
			HARDWARE FAILURE OF ST. #4 LINE 2	5.80E-04	AD4MOD08
			HARDWARE FAILURE OF ST. #4 LINE 3	5.80E-04	AD4MOD09
6	7.13E-11	.36	SAFETY INJECTION LINE BREAK INITIATING EVENT OCCURS	2.12E-04	IEV-SI-LB
			HARDWARE FAILURE OF ST. #4 LINE 1	5.80E-04	AD4MOD07
			HARDWARE FAILURE OF ST. #4 LINE 4	5.80E-04	AD4MOD10
7	7.13E-11	.36	SAFETY INJECTION LINE BREAK INITIATING EVENT OCCURS	2.12E-04	IEV-SI-LB
			HARDWARE FAILURE OF ST. #4 LINE 1	5.80E-04	AD4MOD07
			HARDWARE FAILURE OF ST. #4 LINE 3	5.80E-04	AD4MOD09
8	7.13E-11	.36	SAFETY INJECTION LINE BREAK INITIATING EVENT OCCURS	2.12E-04	IEV-SI-LB
			HARDWARE FAILURE OF ST. #4 LINE 1	5.80E-04	AD4MOD07
			HARDWARE FAILURE OF ST. #4 LINE 2	5.80E-04	AD4MOD08
9	3.65E-11	.18	SAFETY INJECTION LINE BREAK INITIATING EVENT OCCURS	2.12E-04	IEV-SI-LB
			COND. PROB. OF REC-MANDAS (FAILURE OF MANUAL DAS AC	5.06E-01	REC-MANDASC
			OPER. FAILS TO RECOG. THE NEED FOR RCS DEPRESS. DURING MLOCA	3.30E-03	LPM-MAN02
			CCF OF ESF INPUT LOGIC (HARDWARE)	1.03E-04	CCX-INPUT-LOGIC
10	3.34E-11	.17	SAFETY INJECTION LINE BREAK INITIATING EVENT OCCURS	2.12E-04	IEV-SI-LB
			COND. PROB. OF REC-MANDAS (FAILURE OF MANUAL DAS AC	5.06E-01	REC-MANDASC
			OPER. FAILS TO FULFIL MANUAL ACTUATION OF ADS	3.02E-03	ADN-MAN01
			CCF OF ESF INPUT LOGIC (HARDWARE)	1.03E-04	CCX-INPUT-LOGIC

Table 59-7 (Sheet 2 of 3)

SEQUENCE 4 – SAFETY INJECTION LINE BREAK DOMINANT CUTSETS (SI-LB-08)

NUMBER	CUTSET PROB	PERCENT	BASIC EVENT NAME		
11	2.71E-11	.14	SAFETY INJECTION LINE BREAK INITIATING EVENT OCCURS	2.12E-04	IEV-SI-LB
			FAILURE OF MANUAL DAS ACT.	1.16E-02	REC-MANDAS
			CCF OF PMS ESF OUTPUT LOGIC SOFTWARE	1.10E-05	CCX-PMXMOD1-SW
12	2.69E-11	.14	SAFETY INJECTION LINE BREAK INITIATING EVENT OCCURS	2.12E-04	IEV-SI-LB
			COMMON CAUSE FAILURE OF THE BATTERIES IDSA-DB-1A/1B	4.70E-05	CCX-BY-PN
			UNAVAILABILITY OF BUS ECS ES 2 DUE TO UNSCHEDUL MAINTENANCE	2.70E-03	EC2BS002TM
13	2.69E-11	.14	SAFETY INJECTION LINE BREAK INITIATING EVENT OCCURS	2.12E-04	IEV-SI-LB
			COMMON CAUSE FAILURE OF THE BATTERIES IDSA-DB-1A/1B	4.70E-05	CCX-BY-PN
			BUS UNAVAILABLE DUE TO UNSCHEDULED MAINTENANCE	2.70E-03	EC2BS022TM
14	2.69E-11	.14	SAFETY INJECTION LINE BREAK INITIATING EVENT OCCURS	2.12E-04	IEV-SI-LB
			COMMON CAUSE FAILURE OF THE BATTERIES IDSA-DB-1A/1B	4.70E-05	CCX-BY-PN
			BUS UNAVAILABLE DUE TO UNSCHEDULED MAINTENANCE	2.70E-03	EC2BS221TM
15	2.69E-11	.14	SAFETY INJECTION LINE BREAK INITIATING EVENT OCCURS	2.12E-04	IEV-SI-LB
			COMMON CAUSE FAILURE OF THE BATTERIES IDSA-DB-1A/1B	4.70E-05	CCX-BY-PN
			UNAVAILABILITY OF BUS ECS ES 1 DUE TO UNSCHEDULED MAINTENANCE	2.70E-03	EC1BS001TM
16	2.69E-11	.14	SAFETY INJECTION LINE BREAK INITIATING EVENT OCCURS	2.12E-04	IEV-SI-LB
			COMMON CAUSE FAILURE OF THE BATTERIES IDSA-DB-1A/1B	4.70E-05	CCX-BY-PN
			BUS UNAVAILABLE DUE TO UNSCHEDULED MAINTENANCE	2.70E-03	EC1BS012TM
17	2.69E-11	.14	SAFETY INJECTION LINE BREAK INITIATING EVENT OCCURS	2.12E-04	IEV-SI-LB
			COMMON CAUSE FAILURE OF THE BATTERIES IDSA-DB-1A/1B	4.70E-05	CCX-BY-PN
			BUS UNAVAILABLE DUE TO UNSCHEDULED MAINTENANCE	2.70E-03	EC1BS121TM
18	2.33E-11	.12	SAFETY INJECTION LINE BREAK INITIATING EVENT OCCURS	2.12E-04	IEV-SI-LB
			FAILURE OF MANUAL DAS REACTOR TRIP HARDWARE	1.00E-02	MDAS
			CCF OF PMS ESF OUTPUT LOGIC SOFTWARE	1.10E-05	CCX-PMXMOD1-SW
19	2.12E-11	.11	SAFETY INJECTION LINE BREAK INITIATING EVENT OCCURS	2.12E-04	IEV-SI-LB
			FAILURE OF MANUAL DAS ACT.	1.16E-02	REC-MANDAS
			CCF OF EPO BOARDS IN PMS	8.62E-06	CCX-EP-SAM
20	1.91E-11	.10	SAFETY INJECTION LINE BREAK INITIATING EVENT OCCURS	2.12E-04	IEV-SI-LB
			BUS UNAVAILABLE DUE TO TEST OR CORRECTIVE MAINTENANCE	3.00E-04	IDDBSDS1TM
			BUS UNAVAILABLE DUE TO TEST OR CORRECTIVE MAINTENANCE	3.00E-04	IDBBSDS1TM

Table 59-7 (Sheet 3 of 3)

SEQUENCE 4 – SAFETY INJECTION LINE BREAK DOMINANT CUTSETS (SI-LB-08)

NUMBER	CUTSET PROB	PERCENT	BASIC EVENT NAME		
21	1.91E-11	.10	SAFETY INJECTION LINE BREAK INITIATING EVENT OCCURS	2.12E-04	IEV-SI-LB
			BUS UNAVAILABLE DUE TO TEST OR CORRECTIVE MAINTENANCE	3.00E-04	IDDBSDS1TM
			BUS UNAVAILABLE DUE TO TEST OR CORRECTIVE MAINTENANCE	3.00E-04	IDBBSDD1TM
22	1.91E-11	.10	SAFETY INJECTION LINE BREAK INITIATING EVENT OCCURS	2.12E-04	IEV-SI-LB
			BUS UNAVAILABLE DUE TO TEST OR CORRECTIVE MAINTENANCE	3.00E-04	IDDBSDD1TM
			BUS UNAVAILABLE DUE TO TEST OR CORRECTIVE MAINTENANCE	3.00E-04	IDBBSDS1TM
23	1.91E-11	.10	SAFETY INJECTION LINE BREAK INITIATING EVENT OCCURS	2.12E-04	IEV-SI-LB
			BUS UNAVAILABLE DUE TO TEST OR CORRECTIVE MAINTENANCE	3.00E-04	IDDBSDD1TM
			BUS UNAVAILABLE DUE TO TEST OR CORRECTIVE MAINTENANCE	3.00E-04	IDBBSDD1TM
24	1.91E-11	.10	SAFETY INJECTION LINE BREAK INITIATING EVENT OCCURS	2.12E-04	IEV-SI-LB
			BUS UNAVAILABLE DUE TO TEST OR CORRECTIVE MAINTENANCE	3.00E-04	IDCBSDS1TM
			BUS UNAVAILABLE DUE TO TEST OR CORRECTIVE MAINTENANCE	3.00E-04	IDABSDS1TM
25	1.91E-11	.10	SAFETY INJECTION LINE BREAK INITIATING EVENT OCCURS	2.12E-04	IEV-SI-LB
			BUS UNAVAILABLE DUE TO TEST OR CORRECTIVE MAINTENANCE	3.00E-04	IDCBSDS1TM
			BUS UNAVAILABLE DUE TO TEST OR CORRECTIVE MAINTENANCE	3.00E-04	IDABSDD1TM

Table 59-8

SEQUENCE 5 – REACTOR VESSEL RUPTURE CUTSET (RV-RP-02)

NUMBER	CUTSET PROB	PERCENT	BASIC EVENT NAME		
1	1.00E-08	100.00	REACTOR VESSEL RUPTURE INITIATING EVENT OCCURS	1.00E-08	IEV-RV-RP

Table 59-9 (Sheet 1 of 3)

SEQUENCE 6 – SMALL LOCA DOMINANT CUTSETS (SLOCA-05)

NUMBER	CUTSET PROB	PERCENT	BASIC EVENT NAME		
1	6.00E-09	71.10	SMALL LOCA INITIATING EVENT OCCURS	5.00E-04	IEV-SLOCA
			PLUGGING OF BOTH RECIRC LINES DUE TO CCF OF SUMP SCREENS	1.20E-05	REX-FL-GP
2	2.39E-09	28.32	SMALL LOCA INITIATING EVENT OCCURS	5.00E-04	IEV-SLOCA
			CCF OF TANK LEVEL TRANSMITTERS	4.78E-04	IWX-XMTR
			OPER. FAILS TO ACT. SUMP RECIRC GIVEN IRW LEVEL SIGNAL FAILUR	1.00E-02	REN-MAN04
3	2.88E-11	.34	SMALL LOCA INITIATING EVENT OCCURS	5.00E-04	IEV-SLOCA
			SUMP SCREEN A PLUGS AND PREVENTS FLOW	2.40E-04	REA-PLUG
			SUMP SCREEN B PLUGS AND PREVENTS FLOW	2.40E-04	REB-PLUG
4	9.18E-12	.11	SMALL LOCA INITIATING EVENT OCCURS	5.00E-04	IEV-SLOCA
			CCF OF TANK LEVEL TRANSMITTERS	4.78E-04	IWX-XMTR
			CCF OF CMT LEVEL SWITCHES	3.84E-05	CCX-VS-FA
5	2.63E-12	.03	SMALL LOCA INITIATING EVENT OCCURS	5.00E-04	IEV-SLOCA
			CCF OF PMS ESF OUTPUT LOGIC SOFTWARE	1.10E-05	CCX-PMXMOD1-SW
			CCF OF TANK LEVEL TRANSMITTERS	4.78E-04	IWX-XMTR
6	2.63E-12	.03	SMALL LOCA INITIATING EVENT OCCURS	5.00E-04	IEV-SLOCA
			CCX-PMXMOD4-SW	1.10E-05	CCX-PMXMOD4-SW
			CCF OF TANK LEVEL TRANSMITTERS	4 78E-04	IWX-XMTR
7	2.06E-12	.02	SMALL LOCA INITIATING EVENT OCCURS	5.00E-04	IEV-SLOCA
			CCF OF EPO BOARDS IN PMS	8.62E-06	CCX-EP-SAM
			CCF OF TANK LEVEL TRANSMITTERS	4.78E-04	IWX-XMTR
8	3.07E-13	.00	SMALL LOCA INITIATING EVENT OCCURS	5.00E-04	IEV-SLOCA
			HARDWARE FAILURE CAUSE RECIRC. CV 119A FAILS TO OPEN	1.75E-03	REACV119GO
			SUMP SCREEN B PLUGS AND PREVENTS FLOW	2.40E-04	REB-PLUG
			HARDWARE FAILURE OF SQUIB VALVE 118A	1.46E-03	IRWMOD09
9	3.07E-13	.00	SMALL LOCA INITIATING EVENT OCCURS	5.00E-04	IEV-SLOCA
			HARDWARE FAILURE CAUSE RECIRC. CV 119B FAILS TO OPEN	1.75E-03	REBCV119GO
			SUMP SCREEN A PLUGS AND PREVENTS FLOW	2.40E-04	REA-PLUG
			HARDWARE FAILURE OF SQUIB VALVE 118B	1.46E-03	IRWMOD11
10	2.87E-13	.00	SMALL LOCA INITIATING EVENT OCCURS	5.00E-04	IEV-SLOCA
			SOFTWARE CCF OF ALL CARDS	1.20E-06	CCX-SFTW
			CCF OF TANK LEVEL TRANSMITTERS	4.78E-04	IWX-XMTR

Table 59-9 (Sheet 2 of 3)

SEQUENCE 6 – SMALL LOCA DOMINANT CUTSETS (SLOCA-05)

NUMBER	CUTSET PROB	PERCENT	BASIC EVENT NAME		
11	2.56E-13	.00	SMALL LOCA INITIATING EVENT OCCURS	5.00E-04	IEV-SLOCA
			HARDWARE FAILURE OF SQUIB VALVE 120A	1.46E-03	IRWMOD10
			SUMP SCREEN B PLUGS AND PREVENTS FLOW	2.40E-04	REB-PLUG
			HARDWARE FAILURE OF SQUIB VALVE 118A	1.46E-03	IRWMOD09
12	2.56E-13	.00	SMALL LOCA INITIATING EVENT OCCURS	5.00E-04	IEV-SLOCA
			HARDWARE FAILURE OF SQUIB VALVE 120B	1.46E-03	IRWMOD12
			SUMP SCREEN A PLUGS AND PREVENTS FLOW	2.40E-04	REA-PLUG
			HARDWARE FAILURE OF SQUIB VALVE 118B	1.46E-03	IRWMOD11
13	2.39E-13	.00	SMALL LOCA INITIATING EVENT OCCURS	5.00E-04	IEV-SLOCA
			INDICATION FAILURE	1.00E-06	ALL-IND-FAIL
			CCF OF TANK LEVEL TRANSMITTERS	4.78E-04	IWX-XMTR
14	1.84E-13	.00	SMALL LOCA INITIATING EVENT OCCURS	5.00E-04	IEV-SLOCA
			HARDWARE FAILURE CAUSE RECIRC. CV 119A FAILS TO OPEN	1.75E-03	REACV119GO
			SUMP SCREEN B PLUGS AND PREVENTS FLOW	2.40E-04	REB-PLUG
			RELAY FAILS TO OPERATE	8.76E-04	IWBRS118AFA
15	1.84E-13	.00	SMALL LOCA INITIATING EVENT OCCURS	5.00E-04	IEV-SLOCA
			HARDWARE FAILURE CAUSE RECIRC. CV 119B FAILS TO OPEN	1.75E-03	REBCV119GO
			SUMP SCREEN A PLUGS AND PREVENTS FLOW	2.40E-04	REA-PLUG
			RELAY FAILS TO OPERATE	8.76E-04	IWARS118BFA
16	1.68E-13	.00	SMALL LOCA INITIATING EVENT OCCURS	5.00E-04	IEV-SLOCA
			CCF OF 2 OUT 2 LOW PRESSURE RECIRCULATION SQUIB VALVES	5.80E-05	IWX-EV4-SA
			CCF OF MOV 120A AND 120B	5.80E-06	IWX-EV2-SA
17	1.53E-13	.00	SMALL LOCA INITIATING EVENT OCCURS	5.00E-04	IEV-SLOCA
			HARDWARE FAILURE OF SQUIB VALVE 120A	1.46E-03	IRWMOD10
			SUMP SCREEN B PLUGS AND PREVENTS FLOW	2.40E-04	REB-PLUG
			RELAY FAILS TO OPERATE	8.76E-04	IWBRS118AFA
18	1.53E-13	.00	SMALL LOCA INITIATING EVENT OCCURS	5.00E-04	IEV-SLOCA
			HARDWARE FAILURE OF SQUIB VALVE 118A	1.46E-03	IRWMOD09
			SUMP SCREEN B PLUGS AND PREVENTS FLOW	2.40E-04	REB-PLUG
			RELAY FAILS TO OPERATE	8.76E-04	IWDRS120AFA

Table 59-9 (Sheet 3 of 3)

SEQUENCE 6 – SMALL LOCA DOMINANT CUTSETS (SLOCA-05)

NUMBER	CUTSET PROB	PERCENT	BASIC EVENT NAME		
19	1.53E-13	.00	SMALL LOCA INITIATING EVENT OCCURS	5.00E-04	IEV-SLOCA
			HARDWARE FAILURE OF SQUIB VALVE 120B	1.46E-03	IRWMOD12
			SUMP SCREEN A PLUGS AND PREVENTS FLOW	2.40E-04	REA-PLUG
			RELAY FAILS TO OPERATE	8.76E-04	IWARS118BFA
20	1.53E-13	.00	SMALL LOCA INITIATING EVENT OCCURS	5.00E-04	IEV-SLOCA
			HARDWARE FAILURE OF SQUIB VALVE 118B	1.46E-03	IRWMOD11
			SUMP SCREEN A PLUGS AND PREVENTS FLOW	2.40E-04	REA-PLUG
			RELAY FAILS TO OPERATE	8.76E-04	IWCRS120BFA
21	9.21E-14	.00	SMALL LOCA INITIATING EVENT OCCURS	5.00E-04	IEV-SLOCA
			RELAY FAILS TO OPERATE	8.76E-04	IWDRS120AFA
			SUMP SCREEN B PLUGS AND PREVENTS FLOW	2.40E-04	REB-PLUG
			RELAY FAILS TO OPERATE	8.76E-04	IWBRS118AFA
22	9.21E-14	.00	SMALL LOCA INITIATING EVENT OCCURS	5.00E-04	IEV-SLOCA
			RELAY FAILS TO OPERATE	8.76E-04	IWCRS120BFA
			SUMP SCREEN A PLUGS AND PREVENTS FLOW	2.40E-04	REA-PLUG
			RELAY FAILS TO OPERATE	8.76E-04	IWARS118BFA
23	8.88E-14	.00	SMALL LOCA INITIATING EVENT OCCURS	5.00E-04	IEV-SLOCA
			HARDWARE FAILURE CAUSE RECIRC. CV 119B FAILS TO OPEN	1.75E-03	REBCV119GO
			CCF OF 2 OUT 2 LOW PRESSURE RECIRCULATION SQUIB VALVES	5.80E-05	IWX-EV4-SA
			HARDWARE FAILURE CAUSE RECIRC. CV 119A FAILS TO OPEN	1.75E-03	REACV119GO
24	7.41E-14	.00	SMALL LOCA INITIATING EVENT OCCURS	5.00E-04	IEV-SLOCA
			HARDWARE FAILURE CAUSE RECIRC. CV 119B FAILS TO OPEN	1.75E-03	REBCV119GO
			CCF OF 2 OUT 2 LOW PRESSURE RECIRCULATION SQUIB VALVES	5.80E-05	IWX-EV4-SA
			HARDWARE FAILURE OF SQUIB VALVE 120A	1.46E-03	IRWMOD10
25	7.41E-14	.00	SMALL LOCA INITIATING EVENT OCCURS	5.00E-04	IEV-SLOCA
			HARDWARE FAILURE CAUSE RECIRC. CV 119A FAILS TO OPEN	1.75E-03	REACV119GO
			CCF OF 2 OUT 2 LOW PRESSURE RECIRCULATION SQUIB VALVES	5.80E-05	IWX-EV4-SA
			HARDWARE FAILURE OF SQUIB VALVE 120B	1.46E-03	IRWMOD12

Table 59-10 (Sheet 1 of 3)

SEQUENCE 7 – MEDIUM LOCA DOMINANT CUTSETS (MLOCA-05)

NUMBER	CUTSET PROB	PERCENT	BASIC EVENT NAME		
1	5.23E-09	71.13	MEDIUM LOCA INITIATING EVENT OCCURS	4.36E-04	IEV-MLOCA
			PLUGGING OF BOTH RECIRC LINES DUE TO CCF OF SUMP SCREENS	1.20E-05	REX-FL-GP
2	2.08E-09	28.29	MEDIUM LOCA INITIATING EVENT OCCURS	4.36E-04	IEV-MLOCA
			CCF OF TANK LEVEL TRANSMITTERS	4.78E-04	IWX-XMTR
			OPER. FAILS TO ACT. SUMP RECIRC GIVEN IRW LEVEL SIGNAL FAILUR	1.00E-02	REN-MAN04
3	2.51E-11	.34	MEDIUM LOCA INITIATING EVENT OCCURS	4.36E-04	IEV-MLOCA
			SUMP SCREEN A PLUGS AND PREVENTS FLOW	2.40E-04	REA-PLUG
			SUMP SCREEN B PLUGS AND PREVENTS FLOW	2.40E-04	REB-PLUG
4	8.00E-12	.11	MEDIUM LOCA INITIATING EVENT OCCURS	4.36E-04	IEV-MLOCA
			CCF OF TANK LEVEL TRANSMITTERS	4.78E-04	IWX-XMTR
			CCX-VS-FA	3.84E-05	CCX-VS-FA
5	2.29E-12	.03	MEDIUM LOCA INITIATING EVENT OCCURS	4.36E-04	IEV-MLOCA
			CCF OF PMS ESF OUTPUT LOGIC SOFTWARE	1.10E-05	CCX-PMXMOD1-SW
			CCF OF TANK LEVEL TRANSMITTERS	4.78E-04	IWX-XMTR
6	2.29E-12	.03	MEDIUM LOCA INITIATING EVENT OCCURS	4.36E-04	IEV-MLOCA
			CCX-PMXMOD4-SW	1.10E-05	CCX-PMXMOD4-SW
			CCF OF TANK LEVEL TRANSMITTERS	4.78E-04	IWX-XMTR
7	1.80E-12	.02	MEDIUM LOCA INITIATING EVENT OCCURS	4.36E-04	IEV-MLOCA
			CCF OF EPO BOARDS IN PMS	8.62E-06	CCX-EP-SAM
			CCF OF TANK LEVEL TRANSMITTERS	4.78E-04	IWX-XMTR
8	2.67E-13	.00	MEDIUM LOCA INITIATING EVENT OCCURS	4.36E-04	IEV-MLOCA
			HARDWARE FAILURE CAUSE RECIRC. CV 119A FAILS TO OPEN	1.75E-03	REACV119GO
			SUMP SCREEN B PLUGS AND PREVENTS FLOW	2.40E-04	REB-PLUG
			HARDWARE FAILURE OF SQUIB VALVE 118A	1.46E-03	IRWMOD09
9	2.67E-13	.00	MEDIUM LOCA INITIATING EVENT OCCURS	4.36E-04	IEV-MLOCA
			HARDWARE FAILURE CAUSE RECIRC. CV 119B FAILS TO OPEN	1.75E-03	REBCV119GO
			SUMP SCREEN A PLUGS AND PREVENTS FLOW	2.40E-04	REA-PLUG
			HARDWARE FAILURE OF SQUIB VALVE 118B	1.46E-03	IRWMOD11
10	2.50E-13	.00	MEDIUM LOCA INITIATING EVENT OCCURS	4.36E-04	IEV-MLOCA
			SOFTWARE CCF OF ALL CARDS	1.20E-06	CCX-SFTW
			CCF OF TANK LEVEL TRANSMITTERS	4.78E-04	IWX-XMTR

Table 59-10 (Sheet 2 of 3)

SEQUENCE 7 – MEDIUM LOCA DOMINANT CUTSETS (MLOCA-05)

NUMBER	CUTSET PROB	PERCENT	BASIC EVENT NAME		
11	2.23E-13	.00	MEDIUM LOCA INITIATING EVENT OCCURS	4.36E-04	IEV-MLOCA
			HARDWARE FAILURE OF SQUIB VALVE 120A	1.46E-03	IRWMOD10
			SUMP SCREEN B PLUGS AND PREVENTS FLOW	2.40E-04	REB-PLUG
			HARDWARE FAILURE OF SQUIB VALVE 118A	1.46E-03	IRWMOD09
12	2.23E-13	.00	MEDIUM LOCA INITIATING EVENT OCCURS	4.36E-04	IEV-MLOCA
			HARDWARE FAILURE OF SQUIB VALVE 120B	1.46E-03	IRWMOD12
			SUMP SCREEN A PLUGS AND PREVENTS FLOW	2.40E-04	REA-PLUG
			HARDWARE FAILURE OF SQUIB VALVE 118B	1.46E-03	IRWMOD11
13	2.08E-13	.00	MEDIUM LOCA INITIATING EVENT OCCURS	4.36E-04	IEV-MLOCA
			INDICATION FAILURE	1.00E-06	ALL-IND-FAIL
			CCF OF TANK LEVEL TRANSMITTERS	4.78E-04	IWX-XMTR
14	1.60E-13	.00	MEDIUM LOCA INITIATING EVENT OCCURS	4.36E-04	IEV-MLOCA
			HARDWARE FAILURE CAUSE RECIRC. CV 119A FAILS TO OPEN	1.75E-03	REACV119GO
			SUMP SCREEN B PLUGS AND PREVENTS FLOW	2.40E-04	REB-PLUG
			RELAY FAILS TO OPERATE	8.76E-04	IWBRS118AFA
15	1.60E-13	.00	MEDIUM LOCA INITIATING EVENT OCCURS	4.36E-04	IEV-MLOCA
			HARDWARE FAILURE CAUSE RECIRC. CV 119B FAILS TO OPEN	1.75E-03	REBCV119GO
			SUMP SCREEN A PLUGS AND PREVENTS FLOW	2.40E-04	REA-PLUG
			RELAY FAILS TO OPERATE	8.76E-04	IWARS118BFA
16	1.47E-13	.00	MEDIUM LOCA INITIATING EVENT OCCURS	4.36E-04	IEV-MLOCA
			CCF OF 2 OUT 2 LOW PRESSURE RECIRCULATION SQUIB VALVES	5.80E-05	IWX-EV4-SA
			CCF OF MOV 120A AND 120B	5.80E-06	IWX-EV2-SA
17	1.34E-13	.00	MEDIUM LOCA INITIATING EVENT OCCURS	4.36E-04	IEV-MLOCA
			HARDWARE FAILURE OF SQUIB VALVE 120A	1.46E-03	IRWMOD10
			SUMP SCREEN B PLUGS AND PREVENTS FLOW	2.40E-04	REB-PLUG
			RELAY FAILS TO OPERATE	8.76E-04	IWBRS118AFA
18	1.34E-13	.00	MEDIUM LOCA INITIATING EVENT OCCURS	4.36E-04	IEV-MLOCA
			HARDWARE FAILURE OF SQUIB VALVE 118A	1.46E-03	IRWMOD09
			SUMP SCREEN B PLUGS AND PREVENTS FLOW	2.40E-04	REB-PLUG
			RELAY FAILS TO OPERATE	8.76E-04	IWDRS120AFA

Table 59-10 (Sheet 3 of 3)

SEQUENCE 7 – MEDIUM LOCA DOMINANT CUTSETS (MLOCA-05)

NUMBER	CUTSET PROB	PERCENT	BASIC EVENT NAME		
19	1.34E-13	.00	MEDIUM LOCA INITIATING EVENT OCCURS	4.36E-04	IEV-MLOCA
			HARDWARE FAILURE OF SQUIB VALVE 120B	1.46E-03	IRWMOD12
			SUMP SCREEN A PLUGS AND PREVENTS FLOW	2.40E-04	REA-PLUG
			RELAY FAILS TO OPERATE	8.76E-04	IWARS118BFA
20	1.34E-13	.00	MEDIUM LOCA INITIATING EVENT OCCURS	4.36E-04	IEV-MLOCA
			HARDWARE FAILURE OF SQUIB VALVE 118B	1.46E-03	IRWMOD11
			SUMP SCREEN A PLUGS AND PREVENTS FLOW	2.40E-04	REA-PLUG
			RELAY FAILS TO OPERATE	8.76E-04	IWCRS120BFA
21	8.03E-14	.00	MEDIUM LOCA INITIATING EVENT OCCURS	4.36E-04	IEV-MLOCA
			RELAY FAILS TO OPERATE	8.76E-04	IWDRS120AFA
			SUMP SCREEN B PLUGS AND PREVENTS FLOW	2.40E-04	REB-PLUG
			RELAY FAILS TO OPERATE	8.76E-04	IWBRS118AFA
22	8.03E-14	.00	MEDIUM LOCA INITIATING EVENT OCCURS	4.36E-04	IEV-MLOCA
			RELAY FAILS TO OPERATE	8.76E-04	IWCRS120BFA
			SUMP SCREEN A PLUGS AND PREVENTS FLOW	2.40E-04	REA-PLUG
			RELAY FAILS TO OPERATE	8.76E-04	IWARS118BFA
23	7.74E-14	.00	MEDIUM LOCA INITIATING EVENT OCCURS	4.36E-04	IEV-MLOCA
			HARDWARE FAILURE CAUSE RECIRC. CV 119B FAILS TO OPEN	1.75E-03	REBCV119GO
			CCF OF 2 OUT 2 LOW PRESSURE RECIRCULATION SQUIB VALVES	5.80E-05	IWX-EV4-SA
			HARDWARE FAILURE CAUSE RECIRC. CV 119A FAILS TO OPEN	1.75E-03	REACV119GO
24	6.46E-14	.00	MEDIUM LOCA INITIATING EVENT OCCURS	4.36E-04	IEV-MLOCA
			HARDWARE FAILURE CAUSE RECIRC. CV 119B FAILS TO OPEN	1.75E-03	REBCV119GO
			CCF OF 2 OUT 2 LOW PRESSURE RECIRCULATION SQUIB VALVES	5.80E-05	IWX-EV4-SA
			HARDWARE FAILURE OF SQUIB VALVE 120A	1.46E-03	IRWMOD10
25	6.46E-14	.00	MEDIUM LOCA INITIATING EVENT OCCURS	4.36E-04	IEV-MLOCA
			HARDWARE FAILURE CAUSE RECIRC. CV 119A FAILS TO OPEN	1.75E-03	REACV119GO
			CCF OF 2 OUT 2 LOW PRESSURE RECIRCULATION SQUIB VALVES	5.80E-05	IWX-EV4-SA
			HARDWARE FAILURE OF SQUIB VALVE 120B	1.46E-03	IRWMOD12

Table 59-11 (Sheet 1 of 3)

SEQUENCE 8 – SMALL LOCA DOMINANT CUTSETS (SLOCA-12)

NUMBER	CUTSET PROB	PERCENT	BASIC EVENT NAME		
1	4.16E-10	8.14	SMALL LOCA INITIATING EVENT OCCURS	5.00E-04	IEV-SLOCA
			CCF OF 2 SQUIB VALVES TO OPERATE	5.90E-05	ADX-EV-SA2
			MECHANICAL FAILURE OF RNS MOV V055	1.41E-02	RN55MOD1
2	4.16E-10	8.14	SMALL LOCA INITIATING EVENT OCCURS	5.00E-04	IEV-SLOCA
			CCF OF 2 SQUIB VALVES TO OPERATE	5.90E-05	ADX-EV-SA2
			HARDWARE FAILURE OF ISOLATION MOV 011	1.41E-02	RN11MOD3
3	4.16E-10	8.14	SMALL LOCA INITIATING EVENT OCCURS	5.00E-04	IEV-SLOCA
			CCF OF 2 SQUIB VALVES TO OPERATE	5.90E-05	ADX-EV-SA2
			HARDWARE FAILS TO OPEN MOV V022/CB FTC/RELAY FTC	1.41E-02	RN22MOD4
4	4.16E-10	8.14	SMALL LOCA INITIATING EVENT OCCURS	5.00E-04	IEV-SLOCA
			CCF OF 2 SQUIB VALVES TO OPERATE	5.90E-05	ADX-EV-SA2
			HARDWARE FAILS TO OPEN MOV V023/CB FTC/RELAY FTC	1.41E-02	RN23MOD5
5	2.95E-10	5.77	SMALL LOCA INITIATING EVENT OCCURS	5.00E-04	IEV-SLOCA
			CCF OF 2 SQUIB VALVES TO OPERATE	5.90E-05	ADX-EV-SA2
			CASK LOADING PIT UNAVAILABLE DUE TO FUEL UNLOADING OPERATIONS	1.00E-02	CLP-UNAVAILABLE
6	2.11E-10	4.13	SMALL LOCA INITIATING EVENT OCCURS	5.00E-04	IEV-SLOCA
			DUE TO CCF OF 4TH STAGE ADS SQUIB VALVES TO OPERATE	3.00E-05	ADX-EV-SA
			MECHANICAL FAILURE OF RNS MOV V055	1.41E-02	RN55MOD1
7	2.11E-10	4.13	SMALL LOCA INITIATING EVENT OCCURS	5.00E-04	IEV-SLOCA
			DUE TO CCF OF 4TH STAGE ADS SQUIB VALVES TO OPERATE	3.00E-05	ADX-EV-SA
			HARDWARE FAILURE OF ISOLATION MOV 011	1.41E-02	RN11MOD3
8	2.11E-10	4.13	SMALL LOCA INITIATING EVENT OCCURS	5.00E-04	IEV-SLOCA
			DUE TO CCF OF 4TH STAGE ADS SQUIB VALVES TO OPERATE	3.00E-05	ADX-EV-SA
			HARDWARE FAILS TO OPEN MOV V022/CB FTC/RELAY FTC	1.41E-02	RN22MOD4
9	2.11E-10	4.13	SMALL LOCA INITIATING EVENT OCCURS	5.00E-04	IEV-SLOCA
			DUE TO CCF OF 4TH STAGE ADS SQUIB VALVES TO OPERATE	3.00E-05	ADX-EV-SA
			HARDWARE FAILS TO OPEN MOV V023/CB FTC/RELAY FTC	1.41E-02	RN23MOD5
10	1.50E-10	2.93	SMALL LOCA INITIATING EVENT OCCURS	5.00E-04	IEV-SLOCA
			DUE TO CCF OF 4TH STAGE ADS SQUIB VALVES TO OPERATE	3.00E-05	ADX-EV-SA
			CASK LOADING PIT UNAVAILABLE DUE TO FUEL UNLOADING OPERATIONS	1.00E-02	CLP-UNAVAILABLE

Table 59-11 (Sheet 2 of 3)

SEQUENCE 8 – SMALL LOCA DOMINANT CUTSETS (SLOCA-12)

NUMBER	CUTSET PROB	PERCENT	BASIC EVENT NAME		
11	1.45E-10	2.84	SMALL LOCA INITIATING EVENT OCCURS	5.00E-04	IEV-SLOCA
			CCF OF 2 SQUIB VALVES TO OPERATE	5.90E-05	ADX-EV-SA2
			CCF OF STOP CHECK VALVES V015A/B TO OPEN	4.90E-03	RNX-KV1-GO
12	8.55E-11	1.67	SMALL LOCA INITIATING EVENT OCCURS	5.00E-04	IEV-SLOCA
			CCF OF 2 SQUIB VALVES TO OPERATE	5.90E-05	ADX-EV-SA2
			OPERATOR FAILS TO ALIGN AND ACTUATE THE RNS	2.90E-03	RHN-MAN01
13	7.97E-11	1.56	SMALL LOCA INITIATING EVENT OCCURS	5.00E-04	IEV-SLOCA
			CCF OF 2 SQUIB VALVES TO OPERATE	5.90E-05	ADX-EV-SA2
			UNAVAILABILITY OF BUS ECS ES 1 DUE TO UNSCHEDUL MAINTENANCE	2.70E-03	EC1BS001TM
14	7.97E-11	1.56	SMALL LOCA INITIATING EVENT OCCURS	5.00E-04	IEV-SLOCA
			CCF OF 2 SQUIB VALVES TO OPERATE	5.90E-05	ADX-EV-SA2
			BUS UNAVAILABLE DUE TO UNSCHEDULED MAINTENANCE	2.70E-03	EC1BS012TM
15	7.97E-11	1.56	SMALL LOCA INITIATING EVENT OCCURS	5.00E-04	IEV-SLOCA
			CCF OF 2 SQUIB VALVES TO OPERATE	5.90E-05	ADX-EV-SA2
			BUS UNAVAILABLE DUE TO UNSCHEDULED MAINTENANCE	2.70E-03	EC1BS122TM
16	7.58E-11	1.48	SMALL LOCA INITIATING EVENT OCCURS	5.00E-04	IEV-SLOCA
			CCF OF 2 SQUIB VALVES TO OPERATE	5.90E-05	ADX-EV-SA2
			HARDWARE FAILURE OF VALVES ON DVI LINE A (V015A & 017	5.07E-02	RNAMOD09
			HARDWARE FAILURE OF VALVES ON DVI LINE B (V015B & 017	5.07E-02	RNBMOD10
17	7.35E-11	1.44	SMALL LOCA INITIATING EVENT OCCURS	5.00E-04	IEV-SLOCA
			DUE TO CCF OF 4TH STAGE ADS SQUIB VALVES TO OPERATE	3.00E-05	ADX-EV-SA
			CCF OF STOP CHECK VALVES V015A/B TO OPEN	4.90E-03	RNX-KV1-GO
18	6.35E-11	1.24	SMALL LOCA INITIATING EVENT OCCURS	5.00E-04	IEV-SLOCA
			COMMON CAUSE FAILURE OF THE BATTERIES IDSA-DB-1A/1B	4.70E-05	CCX-BY-PN
			UNAVAILABILITY OF BUS ECS ES 2 DUE TO UNSCHEDULED MAINTENANCE	2.70E-03	EC2BS002TM
19	6.35E-11	1.24	SMALL LOCA INITIATING EVENT OCCURS	5.00E-04	IEV-SLOCA
			COMMON CAUSE FAILURE OF THE BATTERIES IDSA-DB-1A/1B	4.70E-05	CCX-BY-PN
			BUS UNAVAILABLE DUE TO UNSCHEDULED MAINTENANCE	2.70E-03	EC2BS022TM
20	6.35E-11	1.24	SMALL LOCA INITIATING EVENT OCCURS	5.00E-04	IEV-SLOCA
			COMMON CAUSE FAILURE OF THE BATTERIES IDSA-DB-1A/1B	4.70E-05	CCX-BY-PN
			BUS UNAVAILABLE DUE TO UNSCHEDULED MAINTENANCE	2.70E-03	EC2BS221TM

Table 59-11 (Sheet 3 of 3)

SEQUENCE 8 – SMALL LOCA DOMINANT CUTSETS (SLOCA-12)

NUMBER	CUTSET PROB	PERCENT	BASIC EVENT NAME		
21	6.35E-11	1.24	SMALL LOCA INITIATING EVENT OCCURS	5.00E-04	IEV-SLOCA
			COMMON CAUSE FAILURE OF THE BATTERIES IDSA-DB-1A/1B	4.70E-05	CCX-BY-PN
			UNAVAILABILITY OF BUS ECS ES 1 DUE TO UNSCHEDULED MAINTENANCE	2.70E-03	EC1BS001TM
22	6.35E-11	1.24	SMALL LOCA INITIATING EVENT OCCURS	5.00E-04	IEV-SLOCA
			COMMON CAUSE FAILURE OF THE BATTERIES IDSA-DB-1A/1B	4.70E-05	CCX-BY-PN
			BUS UNAVAILABLE DUE TO UNSCHEDULED MAINTENANCE	2.70E-03	EC1BS012TM
23	6.35E-11	1.24	SMALL LOCA INITIATING EVENT OCCURS	5.00E-04	IEV-SLOCA
			COMMON CAUSE FAILURE OF THE BATTERIES IDSA-DB-1A/1B	4.70E-05	CCX-BY-PN
			BUS UNAVAILABLE DUE TO UNSCHEDULED MAINTENANCE	2.70E-03	EC1BS121TM
24	5.16E-11	1.01	SMALL LOCA INITIATING EVENT OCCURS	5.00E-04	IEV-SLOCA
			CCF OF 2 SQUIB VALVES TO OPERATE	5.90E-05	ADX-EV-SA2
			CHECK VALVE V013 FAILURE TO OPEN	1.75E-03	RNNCV013GO
25	4.50E-11	.88	SMALL LOCA INITIATING EVENT OCCURS	5.00E-04	IEV-SLOCA
			BUS UNAVAILABLE DUE TO TEST OR CORRECTIVE MAINTENANCE	3.00E-04	IDBBSDS1TM
			BUS UNAVAILABLE DUE TO TEST OR CORRECTIVE MAINTENANCE	3.00E-04	IDDBSDS1TM

Table 59-12 (Sheet 1 of 3)

SEQUENCE 9 – MEDIUM LOCA DOMINANT CUTSETS (MLOCA-12)

NUMBER	CUTSET PROB	PERCENT	BASIC EVENT NAME		
1	3.63E-10	8.14	MEDIUM LOCA INITIATING EVENT OCCURS	4.36E-04	IEV-MLOCA
			CCF OF 2 SQUIB VALVES TO OPERATE	5.90E-05	ADX-EV-SA2
			MECHANICAL FAILURE OF RNS MOV V055	1.41E-02	RN55MOD1
2	3.63E-10	8.14	MEDIUM LOCA INITIATING EVENT OCCURS	4.36E-04	IEV-MLOCA
			CCF OF 2 SQUIB VALVES TO OPERATE	5.90E-05	ADX-EV-SA2
			HARDWARE FAILURE OF ISOLATION MOV 011	1.41E-02	RN11MOD3
3	3.63E-10	8.14	MEDIUM LOCA INITIATING EVENT OCCURS	4.36E-04	IEV-MLOCA
			CCF OF 2 SQUIB VALVES TO OPERATE	5.90E-05	ADX-EV-SA2
			HARDWARE FAILS TO OPEN MOV V022/CB FTC/RELAY FTC	1.41E-02	RN22MOD4
4	3.63E-10	8.14	MEDIUM LOCA INITIATING EVENT OCCURS	4.36E-04	IEV-MLOCA
			CCF OF 2 SQUIB VALVES TO OPERATE	5.90E-05	ADX-EV-SA2
			HARDWARE FAILS TO OPEN MOV V023/CB FTC/RELAY FTC	1.41E-02	RN23MOD5
5	2.57E-10	5.77	MEDIUM LOCA INITIATING EVENT OCCURS	4.36E-04	IEV-MLOCA
			CCF OF 2 SQUIB VALVES TO OPERATE	5.90E-05	ADX-EV-SA2
			CASK LOADING PIT UNAVAILABLE DUE TO FUEL UNLOADING OPERATIONS	1.00E-02	CLP-UNAVAILABLE
6	1.84E-10	4.13	MEDIUM LOCA INITIATING EVENT OCCURS	4.36E-04	IEV-MLOCA
			DUE TO CCF OF 4TH STAGE ADS SQUIB VALVES TO OPERATE	3.00E-05	ADX-EV-SA
			MECHANICAL FAILURE OF RNS MOV V055	1.41E-02	RN55MOD1
7	1.84E-10	4.13	MEDIUM LOCA INITIATING EVENT OCCURS	4.36E-04	IEV-MLOCA
			DUE TO CCF OF 4TH STAGE ADS SQUIB VALVES TO OPERATE	3.00E-05	ADX-EV-SA
			HARDWARE FAILURE OF ISOLATION MOV 011	1.41E-02	RN11MOD3
8	1.84E-10	4.13	MEDIUM LOCA INITIATING EVENT OCCURS	4.36E-04	IEV-MLOCA
			DUE TO CCF OF 4TH STAGE ADS SQUIB VALVES TO OPERATE	3.00E-05	ADX-EV-SA
			HARDWARE FAILS TO OPEN MOV V022/CB FTC/RELAY FTC	1.41E-02	RN22MOD4
9	1.84E-10	4.13	MEDIUM LOCA INITIATING EVENT OCCURS	4.36E-04	IEV-MLOCA
			DUE TO CCF OF 4TH STAGE ADS SQUIB VALVES TO OPERATE	3.00E-05	ADX-EV-SA
			HARDWARE FAILS TO OPEN MOV V023/CB FTC/RELAY FTC	1.41E-02	RN23MOD5
10	1.31E-10	2.94	MEDIUM LOCA INITIATING EVENT OCCURS	4.36E-04	IEV-MLOCA
			DUE TO CCF OF 4TH STAGE ADS SQUIB VALVES TO OPERATE	3.00E-05	ADX-EV-SA
			CASK LOADING PIT UNAVAILABLE DUE TO FUEL UNLOADING OPERATIONS	1.00E-02	CLP-UNAVAILABLE

Table 59-12 (Sheet 2 of 3)

SEQUENCE 9 – MEDIUM LOCA DOMINANT CUTSETS (MLOCA-12)

NUMBER	CUTSET PROB	PERCENT	BASIC EVENT NAME		
11	1.26E-10	2.83	MEDIUM LOCA INITIATING EVENT OCCURS	4.36E-04	IEV-MLOCA
			CCF OF 2 SQUIB VALVES TO OPERATE	5.90E-05	ADX-EV-SA2
			CCF OF STOP CHECK VALVES V015A/B TO OPEN	4.90E-03	RNX-KV1-GO
12	7.46E-11	1.67	MEDIUM LOCA INITIATING EVENT OCCURS	4.36E-04	IEV-MLOCA
			CCF OF 2 SQUIB VALVES TO OPERATE	5.90E-05	ADX-EV-SA2
			OPERATOR FAILS TO ALIGN AND ACTUATE THE RNS	2.90E-03	RHN-MAN01
13	6.95E-11	1.56	MEDIUM LOCA INITIATING EVENT OCCURS	4.36E-04	IEV-MLOCA
			CCF OF 2 SQUIB VALVES TO OPERATE	5.90E-05	ADX-EV-SA2
			UNAVAILABILITY OF BUS ECS ES 1 DUE TO UNSCHEDULED MAINTENANCE	2.70E-03	EC1BS001TM
14	6.95E-11	1.56	MEDIUM LOCA INITIATING EVENT OCCURS	4.36E-04	IEV-MLOCA
			CCF OF 2 SQUIB VALVES TO OPERATE	5.90E-05	ADX-EV-SA2
			BUS UNAVAILABLE DUE TO UNSCHEDULED MAINTENANCE	2.70E-03	EC1BS012TM
15	6.95E-11	1.56	MEDIUM LOCA INITIATING EVENT OCCURS	4.36E-04	IEV-MLOCA
			CCF OF 2 SQUIB VALVES TO OPERATE	5.90E-05	ADX-EV-SA2
			BUS UNAVAILABLE DUE TO UNSCHEDULED MAINTENANCE	2.70E-03	EC1BS122TM
16	6.61E-11	1.48	MEDIUM LOCA INITIATING EVENT OCCURS	4.36E-04	IEV-MLOCA
			CCF OF 2 SQUIB VALVES TO OPERATE	5.90E-05	ADX-EV-SA2
			HARDWARE FAILURE OF VALVES ON DVI LINE A (V015A & 017)	5.07E-02	RNAMOD09
			HARDWARE FAILURE OF VALVES ON DVI LINE B (V015B & 017)	5.07E-02	RNBMOD10
17	6.41E-11	1.44	MEDIUM LOCA INITIATING EVENT OCCURS	4.36E-04	IEV-MLOCA
			DUE TO CCF OF 4TH STAGE ADS SQUIB VALVES TO OPERATE	3.00E-05	ADX-EV-SA
			CCF OF STOP CHECK VALVES V015A/B TO OPEN	4.90E-03	RNX-KV1-GO
18	5.53E-11	1.24	MEDIUM LOCA INITIATING EVENT OCCURS	4.36E-04	IEV-MLOCA
			COMMON CAUSE FAILURE OF THE BATTERIES IDSA-DB-1A/1B	4.70E-05	CCX-BY-PN
			UNAVAILABILITY OF BUS ECS ES 2 DUE TO UNSCHEDULED MAINTENANCE	2.70E-03	EC2BS002TM
19	5.53E-11	1.24	MEDIUM LOCA INITIATING EVENT OCCURS	4.36E-04	IEV-MLOCA
			COMMON CAUSE FAILURE OF THE BATTERIES IDSA-DB-1A/1B	4.70E-05	CCX-BY-PN
			BUS UNAVAILABLE DUE TO UNSCHEDULED MAINTENANCE	2.70E-03	EC2BS022TM
20	5.53E-11	1.24	MEDIUM LOCA INITIATING EVENT OCCURS	4.36E-04	IEV-MLOCA
			COMMON CAUSE FAILURE OF THE BATTERIES IDSA-DB-1A/1B	4.70E-05	CCX-BY-PN
			BUS UNAVAILABLE DUE TO UNSCHEDULED MAINTENANCE	2.70E-03	EC2BS221TM

Table 59-12 (Sheet 3 of 3)

SEQUENCE 9 – MEDIUM LOCA DOMINANT CUTSETS (MLOCA-12)

NUMBER	CUTSET PROB	PERCENT	BASIC EVENT NAME		
21	5.53E-11	1.24	MEDIUM LOCA INITIATING EVENT OCCURS	4.36E-04	IEV-MLOCA
			COMMON CAUSE FAILURE OF THE BATTERIES IDSA-DB-1A/1B	4.70E-05	CCX-BY-PN
			UNAVAILABILITY OF BUS ECS ES 1 DUE TO UNSCHEDULED MAINTENANCE	2.70E-03	EC1BS001TM
22	5.53E-11	1.24	MEDIUM LOCA INITIATING EVENT OCCURS	4.36E-04	IEV-MLOCA
			COMMON CAUSE FAILURE OF THE BATTERIES IDSA-DB-1A/1B	4.70E-05	CCX-BY-PN
			BUS UNAVAILABLE DUE TO UNSCHEDULED MAINTENANCE	2.70E-03	EC1BS012TM
23	5.53E-11	1.24	MEDIUM LOCA INITIATING EVENT OCCURS	4.36E-04	IEV-MLOCA
			COMMON CAUSE FAILURE OF THE BATTERIES IDSA-DB-1A/1B	4.70E-05	CCX-BY-PN
			BUS UNAVAILABLE DUE TO UNSCHEDULED MAINTENANCE	2.70E-03	EC1BS121TM
24	4.50E-11	1.01	MEDIUM LOCA INITIATING EVENT OCCURS	4.36E-04	IEV-MLOCA
			CCF OF 2 SQUIB VALVES TO OPERATE	5.90E-05	ADX-EV-SA2
			CHECK VALVE V013 FAILURE TO OPEN	1.75E-03	RNNCV013GO
25	3.92E-11	.88	MEDIUM LOCA INITIATING EVENT OCCURS	4.36E-04	IEV-MLOCA
			BUS UNAVAILABLE DUE TO TEST OR CORRECTIVE MAINTENANCE	3.00E-04	IDDBSDS1TM
			BUS UNAVAILABLE DUE TO TEST OR CORRECTIVE MAINTENANCE	3.00E-04	IDBBSDS1TM

Table 59-13 (Sheet 1 of 3)

SEQUENCE 10 – SPURIOUS ADS ACTUATION DOMINANT CUTSETS (SPADS-09)

NUMBER	CUTSET PROB	PERCENT	BASIC EVENT NAME		
1	2.75E-09	73.90	SPURIOUS ADS INITIATING EVENT OCCURS	5.40E-05	IEV-SPADS
			COMMON CAUSE FAILURE OF 2 ACCUMULATOR CHECK VALVES	5.10E-05	ACX-CV-GO
2	1.65E-10	4.43	SPURIOUS ADS INITIATING EVENT OCCURS	5.40E-05	IEV-SPADS
			CHECK VALVE 029B FAILS TO OPEN	1.75E-03	ACBCV029GO
			CHECK VALVE 029A FAILS TO OPEN	1.75E-03	ACACV029GO
3	1.65E-10	4.43	SPURIOUS ADS INITIATING EVENT OCCURS	5.40E-05	IEV-SPADS
			CHECK VALVE 029B FAILS TO OPEN	1.75E-03	ACBCV029GO
			CHECK VALVE 028A FAILS TO OPEN	1.75E-03	ACACV028GO
4	1.65E-10	4.43	SPURIOUS ADS INITIATING EVENT OCCURS	5.40E-05	IEV-SPADS
			CHECK VALVE 028B FAILS TO OPEN	1.75E-03	ACBCV028GO
			CHECK VALVE 029A FAILS TO OPEN	1.75E-03	ACACV029GO
5	1.65E-10	4.43	SPURIOUS ADS INITIATING EVENT OCCURS	5.40E-05	IEV-SPADS
			CHECK VALVE 028B FAILS TO OPEN	1 75E-03	ACBCV028GO
			CHECK VALVE 028A FAILS TO OPEN	1.75E-03	ACACV028GO
6	6.87E-11	1.85	SPURIOUS ADS INITIATING EVENT OCCURS	5.40E-05	IEV-SPADS
			FLOW TUNING ORIFICE PLUGS	7.27E-04	ACBOR001SP
			CHECK VALVE 029A FAILS TO OPEN	1.75E-03	ACACV029GO
7	6.87E-11	1.85	SPURIOUS ADS INITIATING EVENT OCCURS	5.40E-05	IEV-SPADS
			FLOW TUNING ORIFICE PLUGS	7.27E-04	ACBOR001SP
			CHECK VALVE 028A FAILS TO OPEN	1.75E-03	ACACV028GO
8	6.87E-11	1.85	SPURIOUS ADS INITIATING EVENT OCCURS	5.40E-05	IEV-SPADS
			CHECK VALVE 029B FAILS TO OPEN	1.75E-03	ACBCV029GO
			FLOW TUNING ORIFICE PLUGS	7.27E-04	ACAOR001SP
9	6.87E-11	1.85	SPURIOUS ADS INITIATING EVENT OCCURS	5.40E-05	IEV-SPADS
			CHECK VALVE 028B FAILS TO OPEN	1.75E-03	ACBCV028GO
			FLOW TUNING ORIFICE PLUGS	7.27E-04	ACAOR001SP
10	2.85E-11	.77	SPURIOUS ADS INITIATING EVENT OCCURS	5.40E-05	IEV-SPADS
			FLOW TUNING ORIFICE PLUGS	7.27E-04	ACBOR001SP
			FLOW TUNING ORIFICE PLUGS	7.27E-04	ACAOR001SP

Table 59-13 (Sheet 2 of 3)

SEQUENCE 10 – SPURIOUS ADS ACTUATION DOMINANT CUTSETS (SPADS-09)

NUMBER	CUTSET PROB	PERCENT	BASIC EVENT NAME		
11	6.48E-12	.17	SPURIOUS ADS INITIATING EVENT OCCURS	5.40E-05	IEV-SPADS
			COMMON CAUSE FAILURE OF ACCUMULATOR TANKS	1.20E-07	ACX-TK-AF
12	2.27E-13	.01	SPURIOUS ADS INITIATING EVENT OCCURS	5.40E-05	IEV-SPADS
			ACCUMULATOR TANK B (T001B) RUPTURES	2.40E-06	ACBTK001AF
			CHECK VALVE 029A FAILS TO OPEN	1.75E-03	ACACV029GO
13	2.27E-13	.01	SPURIOUS ADS INITIATING EVENT OCCURS	5.40E-05	IEV-SPADS
			ACCUMULATOR TANK B (T001B) RUPTURES	2.40E-06	ACBTK001AF
			CHECK VALVE 028A FAILS TO OPEN	1.75E-03	ACACV028GO
14	2.27E-13	.01	SPURIOUS ADS INITIATING EVENT OCCURS	5.40E-05	IEV-SPADS
			CHECK VALVE 029B FAILS TO OPEN	1.75E-03	ACBCV029GO
			ACCUMULATOR TANK A (T001A) RUPTURES	2.40E-06	ACATK001AF
15	2.27E-13	.01	SPURIOUS ADS INITIATING EVENT OCCURS	5.40E-05	IEV-SPADS
			CHECK VALVE 028B FAILS TO OPEN	1.75E-03	ACBCV028GO
			ACCUMULATOR TANK A (T001A) RUPTURES	2.40E-06	ACATK001AF
16	9.42E-14	.00	SPURIOUS ADS INITIATING EVENT OCCURS	5.40E-05	IEV-SPADS
			ACCUMULATOR TANK B (T001B) RUPTURES	2.40E-06	ACBTK001AF
			FLOW TUNING ORIFICE PLUGS	7.27E-04	ACAOR001SP
17	9.42E-14	.00	SPURIOUS ADS INITIATING EVENT OCCURS	5.40E-05	IEV-SPADS
			FLOW TUNING ORIFICE PLUGS	7.27E-04	ACBOR001SP
			ACCUMULATOR TANK A (T001A) RUPTURES	2.40E-06	ACATK001AF
18	6.80E-14	.00	SPURIOUS ADS INITIATING EVENT OCCURS	5.40E-05	IEV-SPADS
			FLOW TUNING ORIFICE RUPTURE	7.20E-07	ACBOR001EB
			CHECK VALVE 029A FAILS TO OPEN	1.75E-03	ACACV029GO
19	6.80E-14	.00	SPURIOUS ADS INITIATING EVENT OCCURS	5.40E-05	IEV-SPADS
			FLOW TUNING ORIFICE RUPTURE	7.20E-07	ACBOR001EB
			CHECK VALVE 028A FAILS TO OPEN	1.75E-03	ACACV028GO
20	6.80E-14	.00	SPURIOUS ADS INITIATING EVENT OCCURS	5.40E-05	IEV-SPADS
			CHECK VALVE 029B FAILS TO OPEN	1.75E-03	ACBCV029GO
			FLOW TUNING ORIFICE RUPTURE	7.20E-07	ACAOR001EB

Table 59-13 (Sheet 3 of 3)

SEQUENCE 10 – SPURIOUS ADS ACTUATION DOMINANT CUTSETS (SPADS-09)

NUMBER	CUTSET PROB	PERCENT	BASIC EVENT NAME		
21	6.80E-14	.00	SPURIOUS ADS INITIATING EVENT OCCURS	5.40E-05	IEV-SPADS
			CHECK VALVE 028B FAILS TO OPEN	1.75E-03	ACBCV028GO
			FLOW TUNING ORIFICE RUPTURE	7.20E-07	ACAOR001EB
22	2.83E-14	.00	SPURIOUS ADS INITIATING EVENT OCCURS	5.40E-05	IEV-SPADS
			FLOW TUNING ORIFICE RUPTURE	7.20E-07	ACBOR001EB
			FLOW TUNING ORIFICE PLUGS	7.27E-04	ACAOR001SP
23	2.83E-14	.00	SPURIOUS ADS INITIATING EVENT OCCURS	5.40E-05	IEV-SPADS
			FLOW TUNING ORIFICE PLUGS	7.27E-04	ACBOR001SP
			FLOW TUNING ORIFICE RUPTURE	7.20E-07	ACAOR001EB

Table 59-14

TYPICAL SYSTEM FAILURE PROBABILITIES, SHOWING HIGHER RELIABILITIES FOR SAFETY SYSTEMS

Failure System/Function	Probability	Fault Tree Name	
CMT Valve Signal	5.7E-07	CMT-IC11	(one train; auto and manual actuation)
PRHR Valve Signal	1.1E-06	RHR-IC01	(one train; auto and manual actuation)
Passive Cont. Cool.	1.8E-06	PCT	
Reactor Trip by PMS	1.2E-05	RTPMS	(including operator actions)
Accumulators	6.9E-05	AC2AB	
IRWST Inj.	6.9E-05	IW2AB	
ADS	9.3E-05	ADS	(including operator actions)
Passive PRHR	2.0E-04	PRT	
Core Makeup Tanks	1.1E-04	CM2SL	
125 vdc 1E Bus	3.1E-04	IDADS1	(one bus only)
DC Bus (Non-1E)	3.4E-04	ED1DS1	(one bus only)
RC Pump Trip	5.9E-04	RCT	
Hydrogen Control	1.0E-01	VLH	
Chilled Water	1.4E-03	VWH	
Containment Isol.	1.6E-03	CIC	
Reactor Trip by DAS	1.7E-03	DAS	(including operator action; excluding MGSET failure))
6900 vac Bus	3.2E-03	ECES1	(one bus only)
CVS	3 4E-03	CVS1	
480 vac Bus	5.9E-03	ECEK11	(one bus only)
Service Water	6.2E-03	SWT	
Comp. Cooling Water	6.3E-03	CCT	
Diesel Generators	1.0E-02	DGEN	
Startup Feedwater	1.7E-02	SFWT	
Compressed Air	1.3E-02	CAIR	
Condenser	2.4E-02	CDS	
Main Feedwater	2.8E-02	FWT	(including condenser)
RNS	9.1E-02	RNR	
Hydrogen Control	1.0E-01	VLH	

Table 59-15

SUMMARY OF AP1000 PRA RESULTS

	Core Damage Frequency		Release Frequency	
	At-Power	**Shutdown**	**At-Power**	**Shutdown**
Internal Events	2.41E-07	1.23E-07	1.95E-08	2.05E-08[2]
Internal Flood	8.8-10[1]	N/A	N/A	N/A
Internal Fire	N/A	N/A	N/A	N/A

Notes:

1. Since the internal fire and internal flooding assessments were conservative, bounding analyses (and the at-power and shutdown analyses were not) it is not appropriate to add these results. That is, the different analyses are not comparable.

2. Estimated as 1.5E-08*1.23E-07/9.0E-08 =2.05E-08 (LRF(AP600)*CDF(AP1000)/CDF(AP600))

N/A = not performed.

Table 59-16

SITE BOUNDARY WHOLE BODY EDE DOSE RISK – 24 HOURS

Release Category	Release Frequency (/reactor year)	Mean Dose (sieverts)	Dose (REM)	Risk (REM/reactor year)	Percent Contribution to Total Risk
CFI	1.89E-10	3.25E+01	3.25E+03	6.14E-07	1.2
CFE	7.47E-09	4.23E+01	4.23E+03	3.16E-05	61.4
IC	2.21E-07	1.82E-02	1.82E+00	4.02E-07	0.8
BP	1.05E-08	1.15E+01	1.15E+03	1.21E-05	23.5
CI	1.33E-09	5.10E+01	5.10E+03	6.78E-06	13.2
CFL	3.45E-13	2.58E+01	2.58E+03	8.90E-10	0.0
	2.4E-07		**Total Risk =**	5.15E-05	100.0

Table 59-17

COMPARISON OF AP1000 PRA RESULTS TO RISK GOALS

Plant/Goal	Core Damage Frequency	Large Release Frequency	Containment Success Probability
Current PWR[(1)]	6.7E-05	5.3E-06	92%
NRC Safety Goal	1E-04	1E-06	90%
AP600	1.7E-07	1.8E-08	89%
AP1000	2 41E-07	1.95E-08	92%

Notes:

1. Selected IPE result (two-loop Westinghouse PWR – internal at-power events and at-power flooding only). Note that there is no shutdown PRA requirement for currently operating plants.

Table 59-18 (Sheet 1 of 24)

AP1000 PRA-BASED INSIGHTS

Insight	Disposition
1. The passive core cooling system (PXS) is composed of the following: - Accumulator subsystem - Core makeup tank (CMT) subsystem - In-containment refueling water storage tank (IRWST) subsystem - Passive residual heat removal (PRHR) subsystem. The automatic depressurization system (ADS), which is part of the reactor coolant system (RCS), also supports passive core cooling functions.	
1a. The accumulators provide a safety-related means of safety injection of borated water to the RCS.	6.3.2
The following are some important aspects of the accumulator subsystem as represented in the PRA:	
- There are two accumulators, each with an injection line to the reactor vessel/direct vessel injection (DVI) nozzle. Each injection line has two check valves in series.	Tier 1 Information
- The reliability of the accumulator subsystem is important. The accumulator subsystem is included in the D-RAP.	17.4
- Diversity between the accumulator check valves and the CMT check valves minimizes the potential for common cause failures.	6.3.2
1b. ADS provides a safety-related means of depressurizing the RCS.	Tier 1 Information
The following are some important aspects of ADS as represented in the PRA:	
ADS has four stages. Each stage is arranged into two separate groups of valves and lines. - Stages 1, 2, and 3 discharge from the top of the pressurizer to the IRWST - Stage 4 discharges from the hot leg to the RCS loop compartment.	Tier 1 Information
Each stage 1, 2, and 3 line contains two motor-operated valves (MOVs).	Tier 1 Information
Each stage 4 line contains an MOV valve and a squib valve.	Tier 1 Information
The valve arrangement and positioning for each stage is designed to reduce spurious actuation of ADS. - Stage 1, 2, and 3 MOVs are normally closed and have separate controls. - Each stage 4 squib valve actuation requires signals from two separate PMS cabinets. - Stage 4 is blocked from opening at high RCS pressures.	6.3.2 & 7.3

Table 59-18 (Sheet 2 of 24)

AP1000 PRA-BASED INSIGHTS

Insight	Disposition
1b. (cont.)	
The ADS valves are automatically and manually actuated via the protection and safety monitoring system (PMS), and manually actuated via the diverse actuation system (DAS).	Tier 1 Information
The ADS valves are powered from Class 1E power.	Tier 1 Information
The ADS valve positions are indicated and alarmed in the control room.	6.3.7
Stage 1, 2, and 3 valves are stroke-tested every cold shutdown. Stage 4 squib valve actuators are tested every 2 years for 20% of the valves.	3.9.6
Because of the potential for counter-current flow limitation in the surgeline, it is essential to establish and maintain venting capability with ADS Stage 4 for gravity injection and containment recirculation following an extended loss of RNS when the RCS is open during shutdown operations.	6.3.3.4.3
ADS 4th stage squib valves receive a signal to open during shutdown conditions using PMS low hot leg level logic.	6.3.3.4.3
The reliability of the ADS is important. The ADS is included in the D-RAP.	17.4
ADS is required by the Technical Specifications to be available in Modes 1 through 6 without the cavity flooded.	16.1
Stages 1, 2, and 3, connected to the top of the pressurizer, provide a vent path to preclude pressurization of the RCS during shutdown conditions if decay heat removal is lost.	16.1
Depressurization of the RCS through ADS minimizes the potential for high-pressure melt ejection events.	
- Procedures will be provided for use of the ADS for depressurization of the RCS after core uncovery.	Emergency Response Guidelines
The ADS mitigates high pressure core damage events which can produce challenges to containment integrity due to the following severe accident phenomena: - High pressure melt ejection - Direct containment heating - Induced steam generator tube rupture - Induced RCS piping rupture and rapid hydrogen release to containment	19.36

Table 59-18 (Sheet 3 of 24)

AP1000 PRA-BASED INSIGHTS

Insight	Disposition
1c. The CMTs provide safety-related means of high-pressure safety injection of borated water to the RCS.	6.3.1
The following are some important aspects of CMT subsystem as represented in the PRA:	
There are two CMTs, each with an injection line to the reactor vessel/DVI nozzle..	6.3.2
- Each CMT has a normally open pressure balance line from an RCS cold leg.	
- Each injection line is isolated with a parallel set of air-operated valves (AOVs).	
- These AOVs open on loss of Class 1E dc power, loss of air, or loss of the signal from the PMS.	
- The injection line for each CMT also has two normally open check valves in series.	
The CMT AOVs are automatically and manually actuated from PMS and DAS.	Tier 1 Information
CMT level instrumentation provides an actuation signal to initiate automatic ADS and provides the actuation signal for the IRWST squib valves to open.	6.3.1 & 7.3.1
The CMT AOV positions are indicated and alarmed in the control room.	6.3.7
CMT AOVs are stroke-tested quarterly.	3.9.6
The CMTs are risk-important for power conditions because the level indicators in the CMTs provide an open signal to ADS and to the IRWST squib valves as the CMTs empty.	
- The CMT subsystem is included in the D-RAP.	17.4
CMT is required by the Technical Specifications to be available in Modes 1 through 5 with RCS pressure boundary intact.	16.1

Table 59-18 (Sheet 4 of 24)

AP1000 PRA-BASED INSIGHTS

	Insight	Disposition
1d.	IRWST subsystem provides a safety-related means of performing the following functions: - Low-pressure safety injection following ADS actuation - Long-term core cooling via containment recirculation - Reactor vessel cooling through the flooding of the reactor cavity by draining the IRWST into the containment. The following are some important aspects of the IRWST subsystem as represented in the PRA:	6.3
	IRWST subsystem has the following flowpaths: - Two (redundant) injection lines from IRWST to reactor vessel/DVI nozzle. Each line is isolated with a parallel set of valves; each set with a check valve in series with a squib valve. - Two (redundant) recirculation lines from the containment to the reactor vessel/DVI injection line. Each recirculation line has two paths: one path contains a squib valve and a MOV, the other path contains a squib valve and a check valve. - The two MOV/squib valve lines also provide the capability to flood the reactor cavity.	Tier 1 Information
	There are screens for each IRWST injection line and recirculation line.	Tier 1 Information
	Squib valves provide the pressure boundary and prevent the check valves from normally seeing a high delta-P.	6.3.3
	Squib valves and MOVs are powered by Class 1E power.	Tier 1 Information
	The squib valves and MOVs for injection and recirculation are automatically and manually actuated via PMS, and manually actuated via DAS.	Tier 1 Information
	The squib valves and MOVs for reactor cavity flooding are manually actuated via PMS and DAS from the control room.	Tier 1 Information
	The injection squib valves and the recirculation squib valves in series with check valves are diverse from the other recirculation squib valves in order to minimize the potential for common cause failure between injection and recirculation/reactor cavity flooding.	6.3.2
	Automatic IRWST injection at shutdown conditions is provided using PMS low hot leg level logic.	6.3.3.4.3 & 7.3.1

Table 59-18 (Sheet 5 of 24)

AP1000 PRA-BASED INSIGHTS

Insight	Disposition
1d. (cont.)	
The positions of the squib valves and MOVs are indicated and alarmed in the control room.	6.3.7
IRWST injection and recirculation check valves are exercised at each refueling. IRWST injection and recirculation squib valve actuators are tested every 2 years for 20% of the valves (This does not require valve actuation). IRWST recirculation MOVs are stroke-tested quarterly.	3.9.6
The reliability of the IRWST subsystem is important. The IRWST subsystem is included in the D-RAP.	17.4
IRWST injection and recirculation are required by Technical Specifications to be available in Modes 1 through 6 without the cavity flooded.	16.1
The operator action to flood the reactor cavity is determined in Emergency Response Guideline AFR-C.1, which instructs the operator to flood the reactor cavity when the core-exit thermocouples reach 1200°F.	Emergency Response Guidelines
PXS recirculation valves are automatically actuated by a low IRWST level signal or manually from the control room, if automatic actuation fails.	6.3
1e. Passive residual heat removal (PRHR) provides a safety-related means of performing the following functions:	6.3.1 & 6.3.3
- Removes core decay heat during accidents	
- Allows automatic termination of RCS leak during a steam generator tube rupture (SGTR) without ADS	
- Allows plant to ride out an ATWS event without rod insertion.	PRA App. A4
The following are some important aspects of the PRHR subsystem as represented in the PRA:	
PRHR is actuated by opening redundant parallel air-operated valves. These air-operated valves open on loss of Class 1E power, loss of air, or loss of the signal from PMS.	6.3.2
The PRHR air-operated valves are automatically actuated and manually actuated from the control room by either PMS or DAS.	Tier 1 Information
Diversity of the PRHR air-operated valves from the CMT air-operated valves minimizes the probability for common cause failure of both PRHR and CMT air-operated valves.	6.3.2

Table 59-18 (Sheet 6 of 24)

AP1000 PRA-BASED INSIGHTS

Insight	Disposition
1e. (cont.)	
Long-term cooling of PRHR will result in steaming to the containment. The steam will normally condense on the containment shell and return to the IRWST by safety-related features. Connections are provided to IRWST from the spent fuel system (SFS) and chemical and volume control system (CVS) to extend PRHR operation. A safety-related makeup connection is also provided from outside the containment through the normal residual heat removal system (RNS) to the IRWST.	6.3.1 & system drawings
Capability exists and guidance is provided for the control room operator to identify a leak in the PRHR HX of 500 gpd. This limit is based on the assumption that a single crack leaking this amount would not lead to a PRHR HX tube rupture under the stress conditions involving the pressure and temperature gradients expected during design basis accidents, which the PRHR HX is designed to mitigate.	6.3.3 & 16.1
The positions of the inlet and outlet PRHR valves are indicated and alarmed in the control room.	6.3.7
PRHR air-operated valves are stroke-tested quarterly. The PRHR HX is tested to detect system performance degradation every 10 years.	3.9.6
PRHR is required by Technical Specifications to be available from Modes 1 through 5 with RCS pressure boundary intact.	16.1
The PRHR HX, in conjunction with the PCS, can provide core cooling for an indefinite period of time. After the IRWST water reaches its saturation temperature, the process of steaming to the containment initiates. Condensation occurs on the steel containment vessel, and the condensate is collected in a safety-related gutter arrangement, which returns the condensate to the IRWST. The gutter normally drains to the containment sump, but when the PRHR HX actuates, safety-related isolation valves in the gutter drain line shut and the gutter overflow returns directly to the IRWST. The following design features provide proper re-alignment for the gutter system valves to direct water to the IRWST:	6.3.2.1.1 & 6.3.7.6
- IRWST gutter and its drain isolation valves are safety-related	
- These isolation valves are designed to fail closed on loss of compressed air, loss of Class 1E dc power, or loss of the PMS signal	
- These isolation valves are actuated automatically by PMS and DAS.	7.3.1.2.7
The PRHR subsystem provides a safety-related means of removing decay heat following loss of RNS cooling during shutdown conditions with the RCS intact.	16.1

Table 59-18 (Sheet 7 of 24)

AP1000 PRA-BASED INSIGHTS

Insight	Disposition
2. The protection and safety monitoring system (PMS) provides a safety-related means of performing the following functions:	Tier 1 Information
- Initiates automatic and manual reactor trip	
- Automatic and manual actuation of engineered safety features (ESF).	
PMS monitors the safety-related functions during and following an accident as required by Regulatory Guide 1.97.	7.1.1
PMS initiates an automatic reactor trip and an automatic actuation of ESF. PMS provides manual initaition of reactor trip. PMS 2-out-of-4 initiation logic reverts to a 2-out-of-3 coincidence logic if one of the 4 channels is bypassed. PMS does not allow simultaneous bypass of 2 redundant channels.	Tier 1 Information
PMS has redundant divisions of safety-related post-accident parameter display.	7.5.2.2.1 & 7.5.4
Each PMS division is powered from its respective Class 1E dc and UPS division.	Tier 1 Information
PMS provides fixed position controls in the control room.	Tier 1 Information
Reliability of the PMS is provided by the following:	
- The reactor trip functions are divided into two subsystems.	7.1.2.1.1
- The ESF functions are processed by two microprocessor-based subsystems that are functionally identical in both hardware and software.	7.1.2.2
Four sensors normally monitor variables used for an ESF actuation. These sensors may monitor the same variable for a reactor trip function.	7.3.1
Continuous automatic PMS system monitoring and failure detection/alarm is provided.	7.1.2
PMS equipment is designed to accommodate a loss of the normal heating, ventilation, and air conditioning (HVAC). PMS equipment is protected by the passive heat sinks upon failure or degradation of the active HVAC.	3.11 & 6.4
The reliability of the PMS is important. The PMS is included in the D-RAP.	17.4
The PMS software is designed, tested, and maintained to be reliable under a controlled verification and validation program written in accordance with IEEE 7-4.3.2 (1993) that has been endorsed by Regulatory Guide 1.152. Elements that contribute to a reliable software design include:	App 1A (Compliance with Reg. Guide 1.152)
- A formalized development, modification, and acceptance process in accordance with an approved software QA plan (paraphrased from IEEE standard, section 5.3, "Quality")	

Table 59-18 (Sheet 8 of 24)

AP1000 PRA-BASED INSIGHTS

Insight	Disposition
2. (cont.)	
- A verification and validation program prepared to confirm the design implemented will function as required (IEEE standard, section 5.3.4, "Verification and Validation")	
- Equipment qualification testing performed to demonstrate that the system will function as required in the environment it is intended to be installed in (IEEE standard, section 5.4, "Equipment Qualification")	
- Design for system integrity (performing its intended safety function) when subjected to all conditions, external or internal, that have significant potential for defeating the safety function (abnormal conditions and events) (IEEE standard, section 5.5, "System Integrity")	
- Software configuration management process (IEEE standard, section 5.3.5, "Software Configuration Management").	
3. The diverse actuation system (DAS) provides a nonsafety-related means of performing the following functions:	Tier 1 Information
- Initiates automatic and manual reactor trip	
- Automatic and manual actuation of selected engineered safety features.	
Diversity is assumed in the PRA that eliminates the potential for common cause failures between PMS and DAS.	
- The DAS automatic actuation signals are generated in a diverse manner from the PMS signals. Diversity between DAS and PMS is achieved by the use of different architecture, different hardware implementations, and different software.	Tier 1 Information
DAS provides control room displays and fixed position controls to allow the operators to take manual actions.	7.7.1
DAS actuates using 2-out-of-2 logic. Actuation signals are output to the loads in the form of normally de-energized, energize-to-actuate signals. The normally de-energized output state, along with the dual 2-out-of-2 redundancy, reduces the probability of inadvertent actuation.	7.7.1.11
The actuation devices of DAS and PMS are capable of independent operation that is not affected by the operation of the other. The DAS is designed to actuate components only in a manner that initiates the safety function.	7.7.1.11
The DAS reactor trip function is to trip the control rods by deenergizing the motor-generator set.	7.7.1.11

Table 59-18 (Sheet 9 of 24)

AP1000 PRA-BASED INSIGHTS

Insight	Disposition
3. (cont.)	
In the PRA it is assumed the following eliminates the potential for common cause failures between automatic and manual DAS functions.	
- DAS manual initiation functions are implemented in a manner that bypasses the signal processing equipment of the DAS automatic logic.	Tier 1 Information
The DAS, including the M-G set field breakers, is included in the D-RAP.	17.4
4. The plant control system (PLS) provides a nonsafety-related means of controlling nonsafety-related equipment.	7.1.3 & 7.7.1
- Automatic and manual control of nonsafety-related functions, including "defense-in-depth" functions.	
- Provides control room indication for monitoring overall plant and nonsafety-related system performance.	
PLS has appropriate redundancy to minimize plant transients.	7.1.3 & 7.7.1.12
PLS provides capability for both automatic control and manual control.	7.1.3
Signal selector algorithmes provide the PLS with the ability to obtain inputs from the PMS. The signal selector algorithms select those protection system signals that represent the actual status of the plant and reject erroneous signals.	7.1.3.2
PLS control functions are distributed across multiple distributed controllers so that single failures within a controller do not degrade the performance of control functions performed by other controllers.	7.1.3.1
5. The onsite power system consists of the main ac power system and the dc power system. The main ac power system is a non-Class 1E system. The dc power system consists of two independent systems: the Class 1E dc system and the non-Class 1E dc system.	
5a. The onsite main ac power system is a non-Class 1E system comprised of a normal, preferred, and standby power supplies.	8.3.1.1
The main ac power system distributes power to the reactor, turbine, and balance of plant auxiliary electrical loads for startup, normal operation, and normal/emergency shutdown.	8.3.1.1.1
The arrangement of the buses permits feeding functionally redundant pumps or groups of loads from separate buses and enhances the plant operational reliability.	8.3.1.1.1

Table 59-18 (Sheet 10 of 24)

AP1000 PRA-BASED INSIGHTS

Insight	Disposition
5a. (cont.)	
During power generation mode, the turbine generator normally supplies electric power to the plant auxiliary loads through the unit auxiliary transformers. During plant startup, shutdown, and maintenance, the main ac power is provided from the high-voltage switchyard. The onsite standby power system powered by the two onsite standby diesel generators supplies power to selected loads in the event of loss of normal and preferred ac power supplies.	8.3.1.1.1
Two onsite standby diesel generator units, each furnished with its own support subsystems, provide power to the selected plant nonsafety-related ac loads.	8.3.1.1.2.1
On loss of power to a 6900 V diesel-backed bus, the associated diesel generator automatically starts and produces ac power. The normal source circuit breaker and bus load circuit breakers are opened, and the generator is connected to the bus. Each generator has an automatic load sequencer to enable controlled loading on the associated buses.	Tier 1 Information
5b. The Class 1E dc and uninterruptible power supply (UPS) system (IDS) provides reliable power for the safety-related equipment required for the plant instrumentation, control, monitoring, and other vital functions needed for shutdown of the plant.	8.3.2.1
There are four independent, Class 1E 125 Vdc divisions. Divisions A and D each consists of one battery bank, one switchboard, and one battery charger. Divisions B and C are each composed of two battery banks, two switchboards, and two battery chargers. The first battery bank in the four divisions is designated as the 24-hour battery bank. The second battery bank in Divisions B and C is designated as the 72-hour battery bank.	Tier 1 Information
The 24-hour battery banks provide power to the loads required for the first 24 hours following an event of loss of all ac power sources concurrent with a design basis accident. The 72-hour battery banks provide power to those loads requiring power for 72 hours following the same event.	Tier 1 Information
Battery chargers are connected to dc switchboard buses. The input ac power for the Class 1E dc battery chargers is supplied from non-Class 1E 480 Vac diesel-generator-backed motor control centers.	8.3.2.1.1.1
The 24-hour and the 72-hour battery banks are housed in ventilated rooms apart from chargers and distribution equipment.	8.3.2.1.3
Each of the four divisions of dc systems are electrically isolated and physically separated to prevent an event from causing the loss of more than one division.	8.3.2.1.3
The Class 1E batteries are included in the D-RAP.	17.4

Table 59-18 (Sheet 11 of 24)

AP1000 PRA-BASED INSIGHTS

	Insight	Disposition
5c.	The non-Class 1E dc and UPS system (EDS) consists of the electric power supply and distribution equipment that provide dc and uninterruptible ac power to nonsafety-related loads.	Tier 1 Information
	The non-Class 1E dc and UPS system consists of two subsystems representing two separate power supply trains.	8.3.2.1.2
	EDS load groups 1, 2, and 3 provide 125 Vdc power to the associated inverter units that supply the ac power to the non-Class 1E uninterruptible power supply ac system.	Tier 1 Information
	The onsite standby diesel-generator-backed 480 Vac distribution system provides the normal ac power to the battery chargers.	Tier 1 Information
	The batteries are sized to supply the system loads for a period of at least two hours after loss of all ac power sources.	8.3.2.1.2
6.	The normal residual heat removal system (RNS) provides a safety-related means of performing the following functions: - Containment isolation for the RNS lines that penetrate the containment. - Isolation of the reactor coolant system at the RNS suction and discharge lines. - Pathway for long-term, post-accident makeup of containment inventory.	Tier 1 Information
	RNS provides a nonsafety-related means of core cooling through: - RCS recirculation cooling during shutdown conditions. - Low pressure pumped makeup flow from the SFS cask loading pit and long-term recirculation from the IRWST and the containment. - Heat removal from IRWST during PRHR operation.	5.4.7
	The RNS has redundant pumps and heat exchangers. The pumps are powered by non-Class 1E power with backup connections from the diesel generators.	5.4.7 & 8.3
	RNS is manually aligned from the control room to perform its core cooling functions. The performance of the RNS is indicated in the control room.	5.4.7
	The RNS containment isolation and pressure boundary valves are safety-related. The motor-operated valves are powered by Class 1E dc power.	Tier 1 Information
	The RNS containment isolation MOVs are automatically and manually actuated via PMS.	7.3.1.2.20

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AP1000 PRA-BASED INSIGHTS

Insight	Disposition
6. (cont.)	
Interfacing system loss-of-coolant accident (LOCA) between the RNS and the RCS is prevented by:	5.4.7.2.2
- Each RNS line is isolated by at least three valves.	
- The RNS equipment outside containment is capable of withstanding the operating pressure of the RCS.	
- The RCS isolation valves are interlocked to prevent their opening at RCS pressures above its design pressure.	
CCS provides cooling to the RNS heat exchanger.	Tier 1 Information
Planned maintenance affecting the RNS cooling function and its support systems CCS and SWS should be performed in modes 1, 2, and 3, when the RNS is not normally operating.	16.3
7. The component cooling water system (CCS) is a nonsafety-related system that removes heat from various components and transfers the heat to the service water system.	Tier 1 Information
The CCS has redundant pumps and heat exchanger.	Tier 1 Information
During normal operation, one CCS pump is operating. The standby pump is aligned to automatically start in case of a failure of the operating CCS pump.	9.2.2.4.2
The CCS pumps are automatically loaded on the standby diesel generator in the event of a loss of normal ac power. The CCS, therefore, continues to provide cooling of required components if normal ac power is lost.	9.2 2.4.5.4
8. The service water system (SWS) is a nonsafety-related system that transfers heat from the component cooling water heat exchangers to the atmosphere.	Tier 1 Information
The SWS has redundant pumps, strainers, and cooling tower cells.	9.2.1.2.1
During normal operation, one SWS train of equipment is operating. The standby train is aligned to automatically start in case of a failure of the operating SWS pump.	9.2.1.2.3.3
The SWS pumps and cooling tower fans are automatically loaded onto their associated diesel bus in the event of a loss of normal ac power. Both pumps and cooling tower fans automatically start after power from the diesel generator is available.	9.2.1.2.3.6

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AP1000 PRA-BASED INSIGHTS

	Insight	Disposition
9.	The chemical and volume control system (CVS) provides a safety-related means to terminate inadvertent RCS boron dilution and to preserve containment integrity by isolation of the CVS lines penetrating the containment.	Tier 1 Information
	The CVS provides a nonsafety-related means to perform the following functions:	Tier 1 Information
	- Makeup water to the RCS during normal plant operation.	
	- Boration following a failure of reactor trip	
	- Makeup water to the pressurizer auxiliary spray line.	
	Two makeup pumps are provided. Each pump provides capability for normal makeup.	9.3.6.3.1
	Two safety-related air-operated valves provide isolation of normal CVS letdown during shutdown operation on low hot leg level.	9.3.6.7
10.	The operation of RNS and its support systems (CCS, SWS, main ac power and onsite power) is RTNSS-important for shutdown decay heat removal during reduced RCS inventory operations.	16.3
	- These systems are included in the D-RAP.	17.4
	Short-term availability controls for the RNS during at-power conditions reduce PRA uncertainties.	16.3
11.	The information used by the COL regarding critical human actions (if any) and risk-important tasks from the PRA, as presented in Chapter 18 of the DCD on human factors engineering, is important in developing and implementing procedures, training, and other human reliability related programs.	18
12.	Sufficient instrumentation and control is provided at the remote shutdown workstation to bring the plant to safe shutdown conditions in case the control room must be evacuated.	7.4.3
	There are no differences between the main control room and remote shutdown workstation controls and monitoring that would be expected to affect safety system redundancy and reliability.	7.4.3.1.1
13.	Separation or protection of the equipment and cabling among the divisions of safety-related equipment and separation of safety-related from nonsafety-related equipment minimizes the probability that a fire or flood would affect more than one safety-related system or train, except in some areas inside containment where equipment will be capable of achieving safe shutdown prior to damage.	3.4.1.1.2 & 9.5.1.1.1, 9.5.1.2.1.1 & 9A
	Although the containment is a single fire area, adequate design features exist for separation (structural or space), suppression, lack of combustibles, or operator action to ensure the plant can achieve safe shutdown.	9A

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AP1000 PRA-BASED INSIGHTS

Insight	Disposition
13. (cont.) To prevent flooding in a radiologically controlled area (RCA) in the Auxiliary Building from propagating to non-radiologically controlled areas, the non-RCAs are separated from the RCAs by 2 and 3-foot walls and floor slabs. In addition, electrical penetrations between RCAs and non-RCAs in the Auxiliary Building are located above the maximum flood level.	3.4.1.2.2.2
14. The following minimizes the probability for fire and flood propagation from one area to another and helps limit risk from internal fires and floods:	
- Fire barriers are sealed, to the extent possible (i.e., doors).	9.5.1.2.1.1
- Structural barriers which function as flood barriers are watertight below the maximum flood level.	3 4.1.1.2
- Establishing administrative controls to maintain the performance of the fire protection system is the responsibility of the COL applicant.	Table 9.5.1-1, Item 29
15. Fire detection and suppression capability is provided in the design. Flooding control features and sump level indication are provided in the design.	3.4.1, 9.5.1.2.1.2, & 9.5.1.8
Establishing administrative controls to maintain the performance of the fire protection system is the responsibility of the COL applicant.	Table 9.5.1-1, Item 29
16. AP1000 main control room fire ignition frequency is limited as a result of the use of low-voltage, low-current equipment and fiber optic cables.	7.1.2 & 7.1.3
There is no cable spreading room in the AP1000 design.	Table 9.5.1-1
17. Redundancy in control room operations is provided within the control room itself for fires in which control room evacuation is not required.	9.5.1.2.1.1
18. The remote shutdown workstation provides redundancy of control and monitoring for safe shutdown functions in the event that main control room evacuation is required.	7.4.3 & 9.5
The remote shutdown workstation is in a fire and flood area separate from the main control room.	3.4.1.2.2.2, 7.1.2, 7.4.3.1.1. & 9A.3.1.2.5
19. Although a main control room fire may defeat manual actuation of equipment from the main control room, it will not affect the automatic functioning of safe shutdown equipment via PMS or manual operation from the remote shutdown workstation. This is because the PMS cabinets, in which the automatic functions are housed, are located in fire areas separate from the main control room.	7.1.2.7 & 9A.3

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AP1000 PRA-BASED INSIGHTS

	Insight	Disposition
20.	The main control room has its own ventilation system, and is pressurized. This prevents smoke, hot gases, or fire suppressants originating in areas outside the control room from entering the control room via the ventilation system.	9.4.1
	There are separate ventilation systems for safety-related equipment divisions (A & C and B & D). This prevents smoke, hot gases, or fire suppressants originating from one fire area to another to the extent that they could adversely affect safe shutdown capabilities.	9.4.1 9.5.1.1.1
	The ventilation system for the remote shutdown workstation is independent of the ventilation system for the main control room.	9.4.1
21.	AP1000 does not rely on ac power sources for safe shutdown capability since the safety-related passive systems do not require ac power sources for operation. Individual fires resulting in loss of offsite power or affecting onsite standby diesel generator operability do not affect safe shutdown capability.	8.1.4.2
22.	Containment isolation functions are not compromised by internal fire or flood. Redundant containment isolation valves in a given line are located in separate fire and flood areas or zones and, if powered, are served by different control and electrical divisions.	6.2.3
	One isolation component in a given line is located inside containment, while the other is located outside containment, and the containment wall is a fire/flood barrier.	6.2.3, 9.5 & 9A
23.	The AP1000 design minimizes potential flooding sources in safety-related equipment areas, to the extent possible. The design also minimizes the number of penetrations through enclosure or barrier walls below the probable maximum flood level. Walls, floors, and penetrations are designed to withstand the maximum anticipated hydrodynamic loads.	3.4.1
24.	The Combined License applicant will confirm the AP1000 certified design will review differences between the as-built plant and the basis for the AP1000 seismic margin analysis.	19.59.10.5
25.	The depressurization of the reactor coolant system below 150 psi facilitates in-vessel retention of molten core debris.	19.36
26.	The reflective reactor vessel insulation provides an engineered flow path to allow the ingression of water and venting of steam for externally cooling the vessel in the event of a severe accident involving core relocation to the lower plenum.	19.39, 5.3.5 & Tier 1 Information
	The reflective insulation panels and support members can withstand pressure differential loading due to the IVR boiling phenomena.	
	Water inlets and steam vents are provided at the entrance and exit of the insulation boundary.	
	The reactor vessel insulation is included in the D-RAP.	17.4

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AP1000 PRA-BASED INSIGHTS

Insight		Disposition
27.	The reactor cavity design provides a reasonable balance between the regulatory requirements for sufficient ex-vessel debris spreading area and the need to quickly submerge the reactor vessel for the in-vessel retention of core debris.	19.39 & Appendix 19B
28.	The design can withstand a best-estimate ex-vessel steam explosion without failing the containment integrity.	Appendix 19B
29.	The containment design incorporates defense-in-depth for mitigating direct containment heating by providing no significant direct flow path for the transport of particulated molten debris from the reactor cavity to the upper containment regions.	Appendix 19B
30.	The hydrogen control system is comprised of passive autocatalytic recombiners (PARs) and hydrogen igniters to limit the concentration of hydrogen in the containment during accidents and beyond design basis accidents, respectively.	Tier 1 Information
	Operability of the hydrogen igniters is addressed by short-term availability controls during modes 1, 2, 5 (with RCS pressure boundary open), and 6 (with upper internals in place or cavity levels less than full).	16.3
	The operator action to activate the igniters is the first step in ERG AFR.C-1 to ensure that the igniter activation occurs prior to rapid cladding oxidation.	Emergency Response Guidelines
31.	Mitigation of the effects of a diffusion flames on the containment shell are addressed by the following containment layout features:	1.2, General Arrangement Drawings
	- Vents from the PXS and CVS compartments (where hydrogen releases can be postulated) to the CMT room are located well away from the containment shell and containment penetrations. The access hatch to the PXS-B compartment is located near the containment wall and is normally closed to adress severe accident considerations. The access hatch to the PXS-B compartment is accessible from Room 11300 on elevation 107'-2".	3.4.1.2.2.1 & 19.41.7
	- IRWST vents are designed so that those located away from the containment wall open to vent hydrogen releases. In this situation IRWST vents located close to the containment wall would not open because flow of hydrogen through the other vents would not result in a IRWST pressure sufficient to open them.	6.2.4.5.1
32.	The containment structure can withstand the pressurization from a LOCA and the global combustion of hydrogen released in-vessel (10 CFR 50.34(f)).	19.41
33.	The steam generator should not be depressurized to cool down the RCS if water is not available to the secondary side. This action protects the tubes from large pressure differential and minimizes the potential for creep rupture. The COL will develop and implement severe accident management guidance using the suggested framework provided in WCAP-13914.	19.59.10

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AP1000 PRA-BASED INSIGHTS

	Insight	Disposition
34.	Depressurizing the RCS and maintaining a water level covering the SG tubes on the secondary side can mitigate fission product releases from a steam generator tube rupture accident. The COL will develop and implement severe accident management guidance using the suggested framework provided in WCAP-13914.	19.59.10
35.	Loss of ac power does not contribute significantly to the core damage frequency. - Nonsafety-related containment spray does not need to be ac independent.	19.59
36.	AP1000 has a nonsafety-related containment spray system.	6.5.2
	Containment spray is not credited in the PRA. Failure of the nonsafety-related containment spray does not prevent the plant achieving the safety goals.	19.59
	The COL will develop and implement severe accident management guidance for operation of the nonsafety-related containment spray system using the suggested framework provided in WCAP-13914.	19.59.10
37.	Passive containment can withstand severe accidents without PCS water cooling the containment shell. Air cooling alone is sufficient to maintain containment pressure below failure pressure with high probability.	19.40
38.	Operation of ADS stage 4 provides a vent path for the severe accident hydrogen to the steam generator compartments, bypassing the IRWST, and mitigating the conditions required to produce a diffusion flame near the containment wall.	19.41
39.	Containment isolation valves controlled by DAS are important in limiting offsite releases following core melt accidents. The containment isolation valves are included in the D-RAP.	17.4
	Operability of DAS for selected containment isolation actuations is addressed by short-term availability controls.	16.3
40.	Reflooding the reactor pressure vessel through the break can have a significant effect on a severe accident by quenching core debris, achieving a controlled stable state, and producing hydrogen.	19.38 & 19.41
41.	The type of concrete used in the basemat is not important.	Appendix 19B
	The reactor cavity design incorporates features that extend the time to basemat melt-through in the event of RPV failure. The cavity design includes: - A minimum floor area of 48 m^2 available for spreading of the molten core debris - A minimum thickness of concrete above the embedded containment liner of 0.85 m	Appendix 19B

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AP1000 PRA-BASED INSIGHTS

Insight	Disposition
41. (cont.) - There is no piping buried in the concrete beneath the reactor cavity; sump drain lines are not enclosed in either of the reactor cavity floor or reactor cavity sump concrete. Thus, there is no direct pathway from the reactor cavity to outside the containment in the event of core-concrete interactions. - The openings between the reactor cavity and cavity sump are small diameter openings in which core debris in the cavity will solidify. Thus, there is no direct pathway for core debris to enter the sump, except in the case where it might spill over the sump curbing.	
42. No safety-related equipment is located outside the Nuclear Island.	1.2 & 3.4.1
43. Capability exists to vent the containment. The COL will develop and implement severe accident management guidance for venting containment using the suggested framework provided in WCAP-13914.	Appendix 19D 19.59.10
44. A list of risk-important systems, structures, and components (SSCs) has been provided in the D-RAP. The risk-significant SSCs are included in the D-RAP.	17.4 17.4
45. The Combined License applicant referencing the AP1000 certified design will review differences between the as-built plant and the design used as the basis for the AP1000 PRA and Table 15.59-29. If the effects of the differences are shown, by a screening analysis, to potentially result in a significant increase in core damage frequency or large release frequency, the PRA will be updated to reflect these differences.	19.59.10
46. There are no watertight doors used for flood protection in the AP1000 design. Plugging of the drain headers is minimized by designing them large enough to accommodate more than the design flow and by making the flow path as straight as possible.	3.4.1.1.2 9.3.5.1.2
47. The maintenance guidelines as described in the Shutdown Evaluation Report (WCAP-14837) should be considered when developing the plant specific operations procedures.	13.5.1
48. Transient combustibles should be controlled.	Table 9.5.1-1, Items 77-83

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AP1000 PRA-BASED INSIGHTS

Insight	Disposition
49. There are two compartments inside containment (PXS-A and PXS-B) containing safe shutdown equipment that normally do not flood although they are below the maximum flood height. Each of these two compartments contains redundant and essentially identical equipment (one accumulator with associated isolation valves as well as isolation valves for one CMT, one IRWST injection line, and one containment recirculation line). A pipe break in one of these compartments can cause that room to flood. These two compartments are physically separated to ensure that a flood in one compartment does not propagate to the other. Drain lines from the PXS-A and PXS-B compartments to the reactor vessel cavity and steam generator compartment are protected from backflow by redundant backflow preventers.	3.4.1.2.2.1
50. There are seven automatically actuated containment isolation valves inside containment subject to flooding. These seven normally closed containment isolation valves would not fail open as a result of the compartment flooding. Also, there is a redundant, normally closed, containment isolation valve located outside containment in series with each of these valves.	3.4.1.2.2.1
51. The passive containment cooling system (PCS) cooling water not evaporated from the vessel wall flows down to the bottom of the containment annulus. Two 100-percent drain openings, located in the side wall of the Shield Building, are always open with screens provided to prevent entry of small animals into the drains.	19.40
52. The major rooms housing divisional cabling and equipment (the battery rooms, dc equipment rooms, I&C rooms, and penetration rooms) are separated by 3-hour fire rated walls. Separate ventilation subsystems are provided for A and C and for B and D division rooms. In order for a fire to propagate from one divisional room to another, it must move past a 3-hour barrier (e.g., a door) into a common corridor and enter the other room through another 3-hour barrier (e.g., another door).	9.5.1 & 9A.3
53. An access bay in the turbine building is provided to protect the north end of the Auxiliary Building, from potential debris produced by a postulated seismic damage of the adjacent Turbine Building.	1.2
54. There are no normally open connections to sources of "unlimited" quantity of water in the electrical and I&C portions of the Auxiliary Building such as that it could affect safe shutdown capabilities.	Figure 9.5.1-1
55. To prevent flooding in a radiologically controlled area (RCA) in the Auxiliary Building from propagating to non-RCAs, the non-RCAs are separated from the RCAs by 2- and 3-foot walls and floor slabs. In addition, electrical penetrations between RCAs and non-RCAs in the Auxiliary Building are located above the maximum flood level.	3.4.1.2.2.2
56. The two 72-hour rated Class 1E division B and C batteries are located above the maximum flood height in the Auxiliary Building considering all possible flooding sources.	3.4.1.2.2.2

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AP1000 PRA-BASED INSIGHTS

Insight	Disposition
57. Flood water in the Turbine Building drains to the yard and does not affect the Auxiliary Building. The presence of watertight walls and floor of the Auxiliary Building valve/penetration room prevents flooding from propagating beyond this area.	3.4.1.2.2.2
58. The mechanical equipment and electrical equipment in the Auxiliary Building are separated to prevent propagation of leaks from the piping and mechanical equipment areas to the Class 1E equipment and Class 1E I&C equipment rooms.	3.4.1.2.2.2
59. Connections to sources of "large" quantity of water are located in the Turbine Building. They are the service water system, which interfaces with the component cooling water system; and the circulating water system, which interfaces with the Turbine Building closed cooling system and the condenser. Features that minimize the flood propagation to other buildings are: - Flow from any postulated ruptures above grade level (elevation 100') in the Turbine Building flows down to grade level via floor grating and stairwells. This grating in the floors also prevents any significant propagation of water to the Auxiliary Building via flow under the doors. - A relief panel in the Turbine Building west wall at grade level directs the water outside the building to the yard and limits the maximum flood level in the Turbine Building to less than 6 inches. Flooding propagation to areas of the adjacent Auxiliary Building, via flow under doors or backflow through the drains, is possible but is bounded by a postulated break in those areas.	3.4.1.2.2.3
60. Flood water in the Annex Building grade level is directed by the sloped floor to drains and to the yard area through the door of the Annex Building. Flow from postulated ruptures above grade level in the Annex Building is directed by floor drains to the Annex Building sump, which discharges to the Turbine Building drain tank. Alternate paths include flow to the Turbine Building via flow under access doors and down to grade level via stairwells and elevator shaft. The floors of the Annex Building are sloped away from the access doors to the Auxiliary Building in the vicinity of the access doors to prevent migration of flood water to the non-RCAs of the Nuclear Island where all safety-related equipment is located.	3.4.1.2.2.3
61. There are no connections to sources of "unlimited" quantity of water, except for fire protection, in the Annex Building.	Figure 9.5.1-1

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AP1000 PRA-BASED INSIGHTS

Insight		Disposition
62.	To prevent overdraining, the RCS hot and cold legs are vertically offset, which permits draining of the steam generators for nozzle dam insertion with a hot leg level much higher than traditional designs.	7.2.1
	To lower the RCS hot leg level at which a vortex occurs in the RNS suction line, a step nozzle connection between the RCS hot leg and the RNS suction line is used.	5.4.7.2.1 & Figure 5.1-5
	Should vortexing occur, air entrainment into the RNS pump suction is limited.	5.4.7.2.1
	There are two safety-related RCS hot leg level channels, one located in each hot leg. These level instruments are independent and do not share instrument lines. These level indicators are provided primarily to monitor RCS level during midloop operations. One level tap is at the bottom of the hot leg, and the other tap is on the top of the hot leg close to the steam generator.	Tier 1 Information Figure 5.1-5 19E.2.1.1
	Wide range pressurizer level indication (cold calibrated) is provided that can measure RCS level to the bottom of the hot legs. This nonsafety-related pressurizer level indication can be used as an alternative way of monitoring level and can be used to identify inconsistencies in the safety-related hot leg level instrumentation.	Tier 1 Information Figure 5.1-5 19E.2.1.1
	The RNS pump suction line is sloped continuously upward from the pump to the reactor coolant system hot leg with no local high points. This design eliminates potential problems in refilling the pump suction line if an RNS pump is stopped when cavitating due to excessive air entrainment. This self-venting suction line allows the RNS pumps to be immediately restarted once an adequate level in the hot leg is re-established.	5.4.7.2.1
	It is important to maximize the availability of the nonsafety-related wide range pressurizer level indication during RCS draining operations during cold shutdown. The Combined License applicant is responsible for developing procedures and training that encompass this item.	13.5
63.	Solid-state switching devices and electro-mechanical relays resistant to relay chatter will be used in the AP1000 safety-related I&C system.	19.55.2.3
64.	The annulus drains will have the same or higher HCLPF value as the Shield Building so that the drain system will not fail at lower acceleration levels causing water blocking of the PCS air baffle.	19.59.10
65.	The ability to close containment hatches and penetrations during Modes 5 & 6 prior to steaming to containment is important. The COL is responsible for developing procedures and training that encompass this item.	13.5 & 16.1
66.	Spurious actuation of squib valves is prevented by the use of a squib valve controller circuit which requires multiple hot shorts for actuation, physical separation of potential hot short locations (e.g., routing of ADS cables in low voltage cable trays, and, in the case of PMS, the use of arm and fire signals from separate PMS cabinets), and provisions for operator action to remove power from the fire zone.	9A.2.7.1

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AP1000 PRA-BASED INSIGHTS

Insight	Disposition
67. For long-term recirculation operation, the RNS pumps can take suction from one of the two sump recirculation lines. Unrestricted flow through both parallel paths is required for success of the sump recirculation function when both RNS pumps are running. If one of the two parallel paths fails to open, operator action is required to manually throttle the RNS discharge valve to prevent pump cavitation.	Emergency Response Guidelines
The containment isolation valves in the RNS piping automatically close via PMS with a high radiation signal. The actuation setpoint was established consistent with a DBA non-mechanistic source term associated with a large LOCA. The containment radiation level for other accidents is expected to be below the point that would cause the RNS MOVs to automatically close.	6.2.3 & 7.3.1.2.20
With the RNS pumps aligned either to the IRWST or the containment sump, the pumps' net positive suction head is adequate to prevent pump cavitation and failure even when the IRWST or sump inventory is saturated.	5.4.7
Emergency response guidelines are provided for aligning the RNS from the control room for RCS injection and recirculation.	Emergency Response Guidelines
The following are additional AP1000 features which contribute to the low likelihood of interfacing system LOCAs between the RNS and the RCS: - A relief valve located in the common RNS discharge line outside containment provides protection against excess pressure. - Two remotely operated MOVs connecting the suction and discharge headers to the IRWST are interlocked with the isolation valves connecting the RNS pumps to the hot leg. This prevents inadvertent opening of these two MOVs when the RNS is aligned for shutdown cooling and potential diversion and draining of reactor coolant system. - Power to the four isolation MOVs connecting the RNS pumps to the RCS hot leg is administratively blocked at their motor control centers during normal power operation.	5.4.7.2
Per the Shutdown Evaluation, operability of the RNS is tested, via connections to the IRWST, before its alignment to the RCS hot leg for shutdown cooling.	19E
Inadvertent opening of RNS valve V024 results in a draindown of RCS inventory to the IRWST and requires gravity injection from the IRWST. The COL applicant is responsible for developing administrative controls to ensure that inadvertent opening of this valve is unlikely.	13.5
The reliability of the IRWST suction isolation valve (V023) to open on demand is important. The IRWST suction isolation valve is included in the D-RAP.	17.4

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AP1000 PRA-BASED INSIGHTS

	Insight	Disposition
68.	The startup feedwater system pumps provide feedwater to the steam generator. This capability provides an alternate core cooling mechanism to the PRHR heat exchangers for non-LOCA or steam generator tube ruptures. The startup feedwater pumps are included in the D-RAP.	17.4
69.	Capability is provided for on-line testing and calibration of the DAS channels, including sensors.	7.7.1.11
	Short-term availability controls of the DAS during at-power conditions reduce PRA uncertainties.	16.3
70.	One CVS pump is configured to operate on demand while the other CVS pump is in standby. The operation of these pumps will alternate periodically.	9.3.6.3.1 & 19.15
	The safety-related PMS boron dilution signal automatically re-aligns CVS pump suction to the boric acid tank. This signal also closes the two safety-related CVS demineralized water supply valves. This signal actuates on reactor trip signal (interlock P-4), source range flux doubling signal, or low input voltage to the Class 1E dc power system battery chargers.	7.3.1.2.14
71.	The COL applicant will maintain procedures to respond to low hot leg level alarms.	Emergency Response Guidelines
72.	The containment recirculation screens are configured such that the chance of clogging is minimized during operation following accidents at power and at shutdown. The configuration features that reduce the chance of clogging include: - Redundant screens are provided and located in separate locations. - Bottom of screens are located well above the lowest containment level as well as the floors around them. - Top of screens are located well below the containment floodup level. - Screens have protective plates that are located close to the top of the screens and extend out in front and to the side of the screens. - Screens have conservative flow areas to account for plugging. Adequate PXS performance can be supported by one screen with at least 90 percent of its surface area completely blocked. - During recirculation operation, the velocities approaching the screens are very low which limits the transport of debris.	6.3.2
73.	A COL applicant cleanliness program controls foreign debris from being introduced into the IRWST tank and into the containment during maintenance and inspection operations.	6.3.2.2.7.2, 6.3.2.2.7.3, & 6.3.8.1

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AP1000 PRA-BASED INSIGHTS

Insight	Disposition
74. For floor drains, from the reactor cavity PXS-A and PXS-B rooms, appropriate precautions such as check valves, back flow preventers, and siphon breaks are assumed to prevent back flow from a flooded space to a nonflooded space.	3.4.1.2.2
75. Plant ventilation systems include features to prevent smoke originating from one fire area to another to the extent that they could adversely affect safe shutdown capabilities.	9.4.2.2
76. An alternative gravity injection path is provided through RNS V-023 during cold shutdown and refueling conditions with the RCS open.	Emergency Response Guidelines
The COL applicant is responsible for developing administrative controls to maximize the likelihood that RNS valve V-023 will be able to open if needed during Mode 5 when the RCS is open, and PRHR cannot be used for core cooling.	13.5
77. The IRWST suction isolation valve (V023) and the RCS pressure boundary isolation valves (V001A/B, V002A/B) are environmentally qualified to perform their safety functions.	Tier 1 Information
78. Following an extended loss of RNS during safe/cold shutdown with the RCS intact and PRHR unavailable, it is essential to establish and maintain venting capability with ADS Stage 4 for gravity injection and containment recirculation.	19.59.5

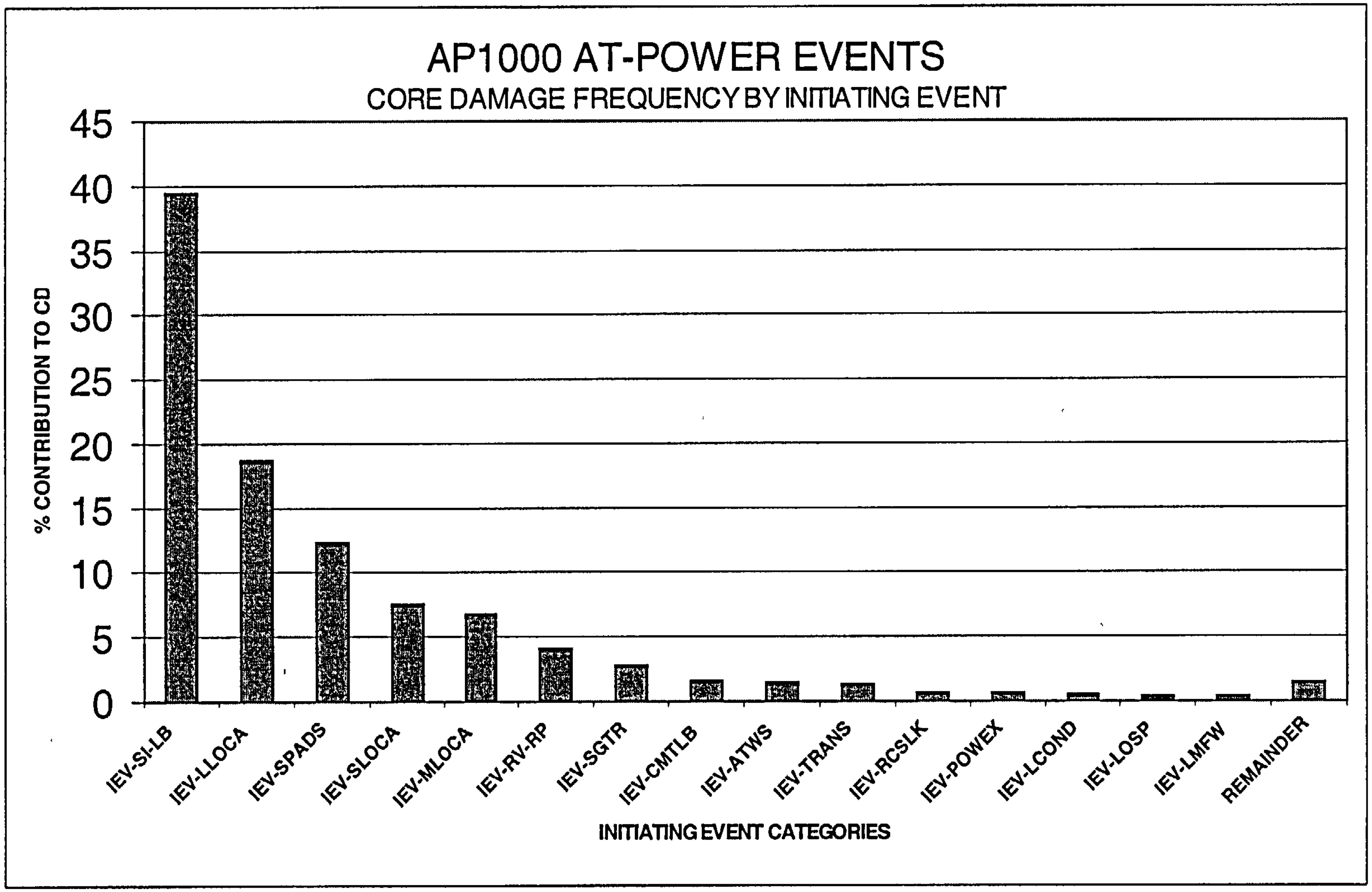

Figure 59-1

Contribution of Initiating Events to Core Damage

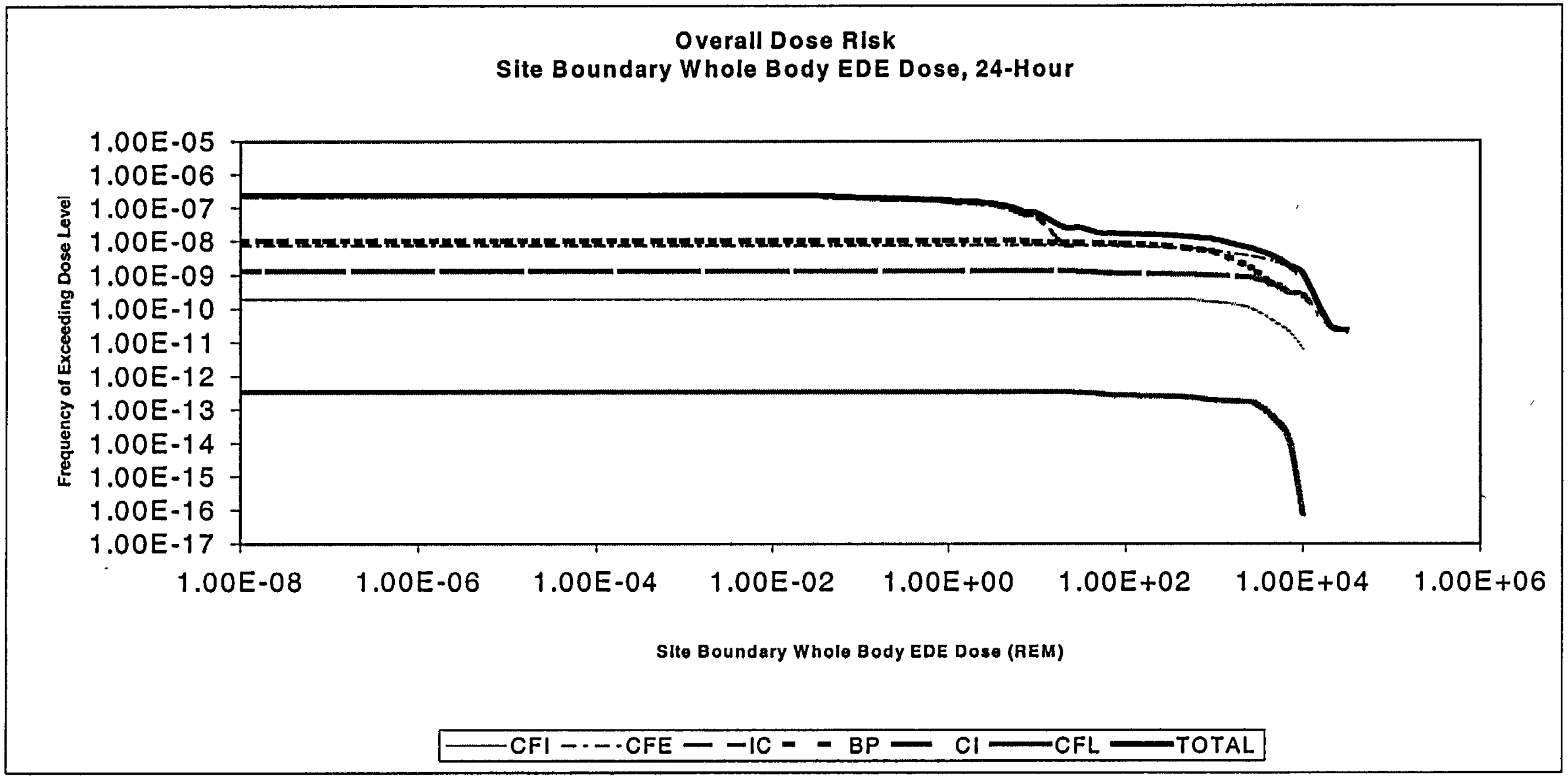

Figure 59-2

24-Hour Site Boundary Dose Cumulative Frequency Distribution

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T of C	i through c	Editorial
Revision 1 Change Roadmap	ci through civ	Editorial
1	1-6	Editorial
3	3-4	Editorial
4B	4B-24	720.027
4B	4B-74	Editorial
6	6-9	720.025 (720.029)
6	6-12	720.025 (720.029) Editorial
6	6-16	Editorial
6	6-17	720.026
6	6-19 through 6-53	720.023 (720.024) (720.097)
6	6-62	720.031
6	6-69	720.025 (720.029)
6	6-71	720.025 (720.029)
8	8-23 and 8-25	Editorial
12	12-6	720.033
12	12-30	720.033
26	26-11 and 26-12	Editorial
28	28-10 and 28-11	Editorial
29	29-17	720.033
29	29-21	720.033

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34	34-1	720.042
34	34-4 and 34-5	720.042
34	34-6 and 34-7	720.042 Editorial
34	34-8 through 34-243	720.042
35	35-28	720.043
39	39-4	720.073 Editorial
39	39-5	720.088 (720.048, 720.074, 720.083, 720.089)
39	39-6	720.088 720.073 (720.048, 720.074, 720.083, and 720.089)
39	39-12	Editorial
39	39-19	Editorial
39	39-20	720.073
39	39-21	Editorial 720.088
39A	39A-1 through 39A-40	720.088 (720.048, 720.074, 720.083, and 720.089)
41	41-6	720.042
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41	41-20	720.054
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43	43-126 and 43-127	Editorial
43C	43C-1 and 43C-2	720.043
44	44-7	Editorial

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45	45-4	470.013 (470.012) Editorial
45	45-5 through 45-48	470.013 (470.012)
49	49-1	720.056
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49	49-9 through 49-49	720.056
50	50-14	720.034
57	57-1 through 57-109	Letter DCP/NRC1515
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57C	57C-1 through 57C-80	Letter DCP/NRC1515
57D	57D-1 through 57D-7	Letter DCP/NRC1515
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59	59-23 through 59-25	Letter DCP/NRC1515
59	59-26	720.056
59	59-28	Editorial
59	59-29 through 59-32	720.038
59	59-34 through 59-38	720.038
59	59-72	Editorial
59	59-73	Technical
59	59-83	Editorial
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59	59-97	720.038 720.066
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59	59-100	Technical
Appendix A	A-1 through A-166	720.007 720.010 720.011 720.012 720.013 720.015 720.016

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Section	Page No.	Type of Change[1]
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Appendix B	B-13	720.076
Appendix D	D-14 and D-15	720.078
Appendix D	D-31	720.078
Appendix D	D-35 and D-36	Technical
Appendix D	D-37 through D-58	720 078

1. Changes incorporated as a result of Westinghouse responses to NRC Request for Additional Information (RAI) identified by RAI number. RAI number in parenthesis contains a reference to RAI response listed above.

APPENDIX A

THERMAL HYDRAULIC ANALYSIS TO SUPPORT SUCCESS CRITERIA

A1 Introduction

The AP1000 design incorporates passive engineered safety features that perform safety-related functions to mitigate accidents and to establish safe shutdown conditions in case nonsafety-related systems are unable to do so. The safety and nonsafety features used in the AP1000 have the same configuration as the AP600 features. In addition, these features have been sized to have similar capabilities as in the AP600. As a result, in most cases the AP1000 features have the same success criteria as the AP600.

The success criteria that define the event tree paths that do not result in core damage have been identified and discussed in PRA Chapter 6. The basis for this success criteria and these success paths are identified in Chapter 6 to be one of the following:

- Plant design calculations
- Plant licensing basis analyses in the *Design Control Document* (DCD)
- Other plant analysis (operating procedures, design transients, etc)
- Plant analysis performed to support PRA success criteria
- Engineering judgment

Section A2 of this document provides an overview of the approach used for the plant analysis performed to support the AP1000 PRA success criteria. This section includes discussion of the approach used for the AP600 PRA success criteria analysis. This section addresses issues that are important to demonstrating successful core cooling for the range of initiating events. A table summary of the AP1000 success criteria is provided. The computer codes used to perform the AP1000 PRA success criteria analysis are discussed.

Section A3 addresses the success criteria that utilize the automatic depressurization system (ADS) in the success criteria. This section discusses the development of the different initiating events including loss-of-coolant accident (LOCA) break sizes. This section is broken down into subsections with cases with automatic ADS and with manual ADS. These subsections include both passive system only mitigation and those utilizing normal residual heat removal system (RNS) pumped injection. Large LOCAs and long term cooling are also addressed.

Section A4 addresses anticipated transients without scram (ATWS).

Section A5 addresses thermal-hydraulic (T/H) uncertainty. An extensive evaluation was performed for the AP600 (Reference A-4) concerning T/H uncertainty. AP1000 makes use of insights from this evaluation and re-analyzes the applicable AP600 T/H uncertainty cases.

A2 Approach to AP1000 PRA Success Criteria

A2.1 Background

For AP600, an extensive range of activities were completed as part of the design and the design certification activities to provide confidence in the design capabilities and reliability of the plant systems and equipment. Special attention was given to the safety-related, passive systems and their associated operating processes. These activities included:

- Incorporation of operational experience (DCD 1.9, 3.1, Appendix 1A, References A-8 and A-9)
- Conservative system design (DCD 6.2, 6.3, 6.4, 7.0, 8.0, References A-5, A-6, and A-7)
- Conservative design basis T/H analysis (DCD 15.0)
- T/H analysis to support PRA success criteria (PRA 6.0 and Appendix A, also References A-3 and A-4)
- Probabilistic risk assessments, including importance and sensitivity studies (PRA Reference A-2)
- Conservative equipment and component design (DCD 3.0, 3.11, ASME codes, ANS standards)
- AP600 plant, system and equipment testing (DCD 1.5, Reference A-10)
- Emergency response guidelines T/H analysis (References A-11 and A-12)
- Plant pre-operational and inservice inspection & testing (DCD 3.9.6, 5.2, 6.6, 16.1, 16.2, 16.3, Reference A-13)

Reference A-14 provides an overview of these activities. For AP600, extensive activities were completed as part of the design certification process to provide confidence in the design capabilities and reliability of the safety-related passive features. To specifically address the multiple-failure accident scenarios that are considered in the PRA, numerous analyses were performed, as documented in the following three reports:

1. PRA success criteria analyses in Appendix A of the PRA (Reference A-2)
2. Benchmarking of MAAP4 to NOTRUMP in WCAP-14869 (Reference A-3)
3. Thermal/hydraulic uncertainty evaluation in WCAP-14800 (Reference A-4)

A large number of accident possibilities are modeled in the PRA, including different sets of operating equipment combined with different initiating events. The accident sequences that lead to successful core cooling were analyzed with the MAAP4 code. MAAP4 was chosen for this task because of its speed, flexibility and ease of use. In addition, some of the design-basis codes were used for selected PRA multiple-failure accident analyses.

The following subsections provide summaries of the analyses that were performed in support of the AP600 PRA, starting with an overview of each of the analysis documents identified above. This is followed by a definition of successful core cooling, and a summary of how the accident sequences are grouped.

A2.2 Acceptance Criteria

Acceptance criteria are the limits established for specific parameters based on known or generally established physical or design limits. Success criteria establish the minimum number or combinations of systems required to operate, during a specified period of time, to ensure that the critical safety functions are met within the limits of the acceptance criteria.

Meeting the success and acceptance criteria ensures that the following critical safety functions are met:

- Decay heat removal (core cooling)
- Reactor coolant system (RCS) inventory control
- RCS pressure control
- Containment heat removal and containment isolation
- Reactivity control

The acceptance criteria are stated in the following paragraphs, which also provide the bases for their implementation.

Decay Heat Removal (Core Cooling) and RCS Inventory Control

Adequacy of core cooling is established by requiring that either the core remains covered with water or the peak cladding temperature (PCT) of the fuel is less than 2200°F at all times during an event. In addition, small core uncovery that that have an extended and slow recovery are not considered success even if the PCT is below 2200°F.

RCS Pressure Control

Adequacy of pressure control is established by requiring that the peak RCS pressure does not exceed the pressure limit corresponding to the service limit stress of the ASME Boiler and Pressure Vessel Code for Level C ("emergency condition") events. As applied to the RCS as a system, this limit denotes the pressure limit for the lowest-rated RCS component. The appropriate pressure limit for Westinghouse PWRs is 3200 psig. That limit corresponded to RCS components other than the reactor vessel (i.e., the reactor vessel limit was substantially higher than 3200 psig). This is judged to be the case for the AP1000 design as well (i.e., the vessel will not be limiting). Therefore, the 3200 psig pressure limit is judged to be applicable.

Containment Heat Removal and Containment Isolation

Adequacy of containment heat removal is established by requiring that the peak containment pressure remain below the ultimate containment pressure. This is done by passive containment heat removal, without the need for the passive containment heat removal water system (PCS). Containment isolation is not necessary if sufficient water is retained in the containment for an extended period of time due to operation of passive containment heat removal.

Reactivity Control

The reactivity control requirement is that subcriticality be rapidly achieved and subsequently maintained. Methods of accomplishing this are event-specific, but generally include one or more of the following: reactor trip with insertion of most of the rod cluster control assemblies (for all events except large LOCA and spurious ADS); termination of any ongoing excessive cooldown or boron dilution, and/or addition of additional boron (for excessive cooldown, e.g., steamline break, or boron dilution events); existence of sufficient voiding in the core coolant to cause initial shutdown, with subsequent addition of boron (for large LOCAs).

Core damage is assumed if any of the following occurs:

- Core cooling acceptance criteria are not met
- RCS pressure control acceptance criteria are not met
- Reactivity control cannot be achieved

A2.3 Summary of AP1000 Success Criteria

PRA Chapter 6 lists all of the AP1000 success criteria. The success criteria are summarized below.

Event	Success Criteria
Transients, Loss of Offsite Power, Loss Cooling Water System	- 1/3 MFW pumps feeding 1/2 steam generator (SG), water supply from condenser via 1/3 condensate pumps, turbine bypass valves to condenser with condenser cooling
	OR
	- 1/2 startup feedwater system (SFW) pumps feeding 1/2 SG, SG steaming via turbine bypass valves to condenser with condenser cooling **OR** SG power-operated relief valve (PORV) (1/SG) **OR** SG safety valves (1/SG)
	OR
	- Passive residual heat removal heat exchanger (PRHR HX)
	OR
	- Feed-Bleed RCS cooling (any small LOCA success)

Event		Success Criteria (continued)
ATWS	-	1/2 SFW pumps feeding 2/2 SGs with SG steaming via turbine bypass valves to condenser with condenser cooling **OR** SG PORV (1/SG) **OR** SG safety valves (1/SG), 1/2 chemical and volume control system (CVS) pumps **OR** 1/2 core makeup tank (CMT) **AND** any small LOCA success
	OR	
	-	PRHR HX, 1/2 CVS pumps **OR** 1/2 CMT **AND** any small LOCA success
Steam Line Break (downstream of MSIV)	-	Close 2/2 main steam isolation valve (MSIV), any transient success
	OR	
	-	1/2 CMT **OR** 1/2 CVS makeup pumps, any transient success
Steam Line Break (upstream of MSIV)	-	1/2 CMT **OR** 1/2 CVS makeup pumps, any transient success
Steam Generator Tube Rupture (SGTR)	-	1/2 CVS makeup pumps, 1/2 SFW pumps feeding the intact SG with SG steaming via turbine bypass valves **OR** intact SG PORV, faulted SG isolation, RCS depressurization (normal pressurizer spray, CVS auxiliary spray **OR** 1/2 ADS stage 1 partial open/close)
	OR	
	-	PRHR HX, 1/2 CMT, isolation all SGs
	OR	
	-	Feed-bleed RCS cooling (any small LOCA success)
RCS Leak	-	1/2 CVS makeup pumps, any transient success
	OR	
	-	Any small LOCA success
PRHR HX Tube Rupture	-	1/2 CVS makeup pumps, close 1/1 PRHR HX inlet motor-operated valve (MOV)
	OR	
	-	Any small LOCA success

Event	Success Criteria (continued)
Small LOCA	- Full ADS [1], 1/2 CMT, 1/2 in-containment refueling water storage tank (IRWST) injection lines, 1/4 containment recirculation valves with containment isolation **OR** 2/4 containment recirculation valves without containment isolation [3]
	OR
	- Full ADS [1], 1/2 accumulator, 1/2 IRWST injection lines, 1/4 containment recirculation valves with containment isolation **OR** 2/4 containment recirculation valves without containment isolation [3]
	OR
	- Part ADS [2], 1/2 CMT, 1/2 RNS pumps, 1/4 containment recirculation valves [3]
	OR
	- Part ADS [2], 1/2 accumulator, 1/2 RNS pumps, 1/4 containment recirculation valves [3]
Medium LOCA Break	- Full ADS [1], 1/2 CMT, 1/2 IRWST injection lines, 1/4 containment recirculation valves with containment isolation **OR** 2/4 containment recirculation valves without containment isolation [3]
	OR
	- Full ADS [1], PRHR HX, 1/2 accumulator, 1/2 IRWST injection lines, 1/4 containment recirculation valves with containment isolation **OR** 2/4 containment recirculation valves without containment isolation [3]
	OR
	- Part ADS [2], 1/2 CMT, 1/2 RNS pumps, 1/4 containment recirculation valves [3]
	OR
	- Part ADS [2], PRHR HX, 1/2 accumulator, 1/2 RNS pumps, 1/4 containment recirculation valves [3]

Event	Success Criteria (continued)
CMT Balance Line Break	- Full ADS [1], 1/1 CMT, 1/2 IRWST injection lines, 1/4 containment recirculation valves with containment isolation **OR** 2/4 containment recirculation valves without containment isolation [3]
	OR
	- Full ADS [1], PRHR HX, 1/2 accumulator, 1/2 IRWST injection lines, 1/4 containment recirculation valves with containment isolation **OR** 2/4 containment recirculation valves without containment isolation [3]
	OR
	- Part ADS [2], 1/1 CMT, 1/2 RNS pumps, 1/4 containment recirculation valves [3]
	OR
	- Part ADS [2], PRHR HX, 1/2 accumulator, 1/2 RNS pumps, 1/4 containment recirculation valves [3]
DVI Break	- Full ADS [1], 1/1 CMT, 1/1 IRWST injection lines, 1/4 containment recirculation valves with containment isolation **OR** 2/4 containment recirculation valves without containment isolation [3]
	OR
	- Full ADS [1], PRHR HX, 1/1 accumulator, 1/1 IRWST injection lines, 1/4 containment recirculation valves with containment isolation **OR** 2/4 containment recirculation valves without containment isolation [3]
Spurious ADS	- Full ADS [1], 1/2 accumulator, 1/2 CMT, 1/2 IRWST injection lines, 1/4 containment recirculation valves with containment isolation **OR** 2/4 containment recirculation valves without containment isolation [3]

Event		Success Criteria (continued)
Large LOCA	-	Full ADS [1], 2/2 accumulator, 1/2 CMT, 1/1 IRWST injection lines, 1/4 containment recirculation valves with containment isolation **OR** 2/4 containment recirculation valves without containment isolation [3]

Notes:
(1) See Table A2.3-1 for detailed full ADS (support passive injection/recirculation) success criteria.
(2) See Table A2.3-2 for detailed partial ADS (support RNS injection/recirculation) success criteria.
(3) Each IRWST line has redundant/parallel valves. Each containment recirculation line has redundant/parallel valves. Note that either containment recirculation line can support either IRWST line by using the bottom of IRWST as a cross connection. This capability also works in the case where there is a break in a direct vessel injection (DVI) line, because when recirculation begins the containment/passive core cooling system (PXS) valve room water level is high enough to provide RCS injection.

A.2.4 Computer Codes Use in PRA Success Criteria Analysis

A.2.4.1 General

As discussed in Section A2.1, there are different methods that are used to verify the success criteria for the AP1000. These methods include design calculations, DCD safety analysis, other transient analysis, and PRA success criteria analysis. For the PRA specific analysis that Westinghouse has performed for the AP1000 to determine the PRA success criteria and to bound the T/H uncertainty, the following computer codes have been used:

MAAP4	-	Post ADS success criteria, both short term and long term core cooling
LOFTRAN	-	ATWS success criteria
NOTRUMP	-	T/H uncertainty analysis for short term core cooling following small LOCAs
LOCTA	-	T/H uncertainty analysis for short term core cooling cladding temperatures
WCOBRA-TRAC	-	T/H uncertainty analysis for long term core cooling

Except for MAAP4, these computer codes have been validated and approved for DCD Chapter 15, "Accident Analysis." Section A5 discusses the use of the NOTRUMP, LOCTA and WCOBRA-TRAC to bound the AP1000 T/H uncertainty.

A.2.4.2 Use of MAAP4

MAAP4 is a computer code that simulates the response of light water reactor systems to initiating events. It was originally developed to investigate the physical phenomena that may occur in the event of a severe accident after significant core damage. Although the emphasis in the code development has been on the severe fuel damage phase of the accident, the code can also be used to determine the thermal-hydraulic behavior prior to core damage.

MAAP4 is a fully integrated, systems accident code and includes models for important thermal-hydraulic and fission-product phenomena which may occur during a postulated accident in a pressurized water reactor (PWR) plant. The models in MAAP4 relevant to success criteria are the following:

- Reactor coolant system thermal-hydraulics
- Cladding water reaction
- Reactor core heatup
- Containment thermal-hydraulics

The version of MAAP4 used for these analyses is documented in Reference A-19, which provides details of the code models, the non-AP600 benchmarking performed, and users guidance.

MAAP4 was used to determine the AP600 PRA success criteria because of its capability to analyze the reactor, passive safety-related systems, active nonsafety-related systems and the containment in an integrated fashion.

MAAP4 was benchmarked for its use for AP600, as documented in Reference A-3, against the more detailed models in NOTRUMP, the Westinghouse-validated code for AP600 small-break LOCAs. A total of 19 benchmarking cases were analyzed with both MAAP4 and NOTRUMP. The first 7 cases were chosen at limiting break sizes across the spectrum of the break sizes analyzed with MAAP4. They demonstrate the basic phenomena that were identified in the PRA Phenomena Identification Ranking Tables (PIRTs), also documented in Reference A-2. The remaining benchmarking cases were sensitivities to demonstrate the capability of MAAP4 to predict trends for different break locations, different number of core make-up tanks or accumulators, different number of automatic depressurization system lines, and different parameters affecting IRWST gravity injection.

The benchmarking work not only provides clear definitions of MAAP4 capabilities and limitations, it provides information on the response of the AP600 plant to multiple failure accidents. The response of the plant is based not only on MAAP4 calculations, but on NOTRUMP analyses. Many of the benchmarking cases are defined based on the PRA success criteria, which means that the least required equipment is credited to show that successful core cooling is achieved.

In Reference A-15, the NRC summarizes the MAAP4/NOTRUMP benchmarking work as follows: "The staff reviewed WCAP-14869 and evaluated Westinghouse's conclusions regarding the adequacy of MAAP4 for screening PRA sequences. The staff found that, in most cases, MAAP4 and NOTRUMP predicted similar trends for system behavior in the base cases and sensitivity analyze. On the basis of the benchmark study comparisons, the staff has determined that MAAP4 is an adequate screening tool for evaluating PRA success criteria for the AP600, subject to the limitations discussed by Westinghouse in WCAP-14869."

A3 Success Criteria Utilizing ADS

A3.1 Grouping of Success Paths With ADS Actuation

In the PRA, LOCAs are sub-divided into different initiating event categories based primarily on the break size, and sometimes the break location. The break location is considered separately if it affects the equipment that may be available to mitigate the event. Transient events (non-LOCAs) are considered separately based on the initiating equipment failure. The different initiating events for LOCAs and transients are defined and modeled in individual event trees. An event tree contains paths of accidents with different sets of equipment failures and successes considered. Sets of equipment successes that lead to successful core cooling are defined as "success criteria."

The initiating events and success paths are listed in Table A3.1-1 and Table A3.1-2, respectively, for the accident sequences that include ADS actuation as a part of successful core cooling. The similarities and differences in the plant response for the different initiating events and types of success paths are discussed below.

Initiating Events

Table A3.1-1 identifies 23 initiating events that may include ADS actuation as a part of the accident sequence that leads to successful core cooling. The initiating events can be lumped into the following groupings to further analyze and describe the plant response:

1. Large LOCA (LLOCA)
2. Medium LOCAs (MLOCA)
3. Small LOCAs (SLOCAs) and high pressure events

Table A3.1-1 summarizes the basis for the size definition for LOCA used in the AP1000 PRA. LLOCAs are defined as a primary system break sufficiently large such that injection from both accumulators are required. Note that because the DVI lines are relatively small compared to the main RCS loop piping, the break of a DVI line is not classified as an LLOCA. As a result, an LLOCA cannot cause the spill of an accumulator or a CMT. Operation of the ADS valves are not required in order to depressurize the RCS to the RNS injection pressure. Operation of ADS valves is required in order to depressurize the RCS to allow gravity injection from the IRWST and containment recirculation. Because of the large size of these LOCAs, the accumulators will empty in as short a time as 3 minutes. This short time does not provide sufficient time for operator action to open the ADS valves and the IRWST injection isolation valves; as a result a CMT is required to provide automatic signals for these valves. The corresponding break size is a break with an equivalent inside diameter of approximately 9 inches or larger.

A special category of LLOCAs is spurious ADS. The opening of all four ADS 4 paths bounds this LOCA. Although the break area of such an event is within the LLOCA size, it only requires one accumulator because of the less severe plant response to the hot leg (HL) LOCA location. Otherwise the mitigating system requirements are the same as for a LLOCA.

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 (SG) 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 progression does not lead to ADS actuation. However, if decay heat removal is not provided, the RCS will eventually pressurize until the pressurizer safety valve setpoint is reached. The resulting loss of coolant through the pressurizer safety valve will cause the accident to proceed as the other high pressure events requiring ADS.

The similarity of the plant response for these high pressure events was shown for AP600 in References A-2 and A-3. Thus, for AP1000, representative high pressure sequences are used to illustrate the plant response to different types of success paths with ADS actuation.

Success Paths

For each event tree that includes ADS actuation as a necessary part of successful core cooling, the success paths with ADS actuation can be categorized depending on whether "automatic" or "manual" ADS actuation is modeled, and whether "full" or "partial" ADS is credited. The automatic and manual ADS scenarios are generally linked to whether 1 CMT or 1 accumulator is credited, and the full and partial ADS are generally linked to whether IRWST gravity injection of RNS pumped injection occurs. Table A3.1-2 lists the success paths that are associated with each of the four main types of ADS success sequences:

- Automatic, full ADS (CMT, IRWST gravity injection)
- Automatic, partial ADS (CMT, RNS pumped injection)
- Manual, full ADS (Accumulator, IRWST gravity injection)
- Manual, partial ADS (Accumulator, RNS pumped injection)

The automatic actuation of ADS is based on low CMT level signals, and thus automatic ADS actuation is credited when at least 1 CMT successfully injects. Full ADS is usually defined as 3 out of 4 stage 4 ADS valves opening, which provides sufficient depressurization of the RCS to achieve gravity injection from the IRWST. IRWST gravity injection typically requires that the RCS pressure be reduced to within approximately 15 psi of the containment pressure. Success paths with automatic ADS actuation leading to IRWST gravity injection are discussed in Section A3.2.1.

If full ADS actuation fails (which means that less than 3 stage 4 ADS valves open), then the event trees consider the possibility that partial ADS actuation may lead to successful core cooling. Partial ADS is the opening of at least 2 stage 2 or stage 3 ADS valves or 1 stage 4 valve, which will depressurize the RCS enough to allow pumped RNS injection when the RCS pressure is less than approximately 175 psia. Success paths with automatic ADS actuation leading to RNS pumped injection are discussed in Section A3.2.2.

If ADS actuation does not automatically actuate on a low CMT level signal, the event trees also model the possibility of manual ADS actuation. This occurs in success paths with the failure of all CMTs. When neither CMT is injecting, the success paths include the need for at least 1 accumulator to inject to the RCS. Because a low CMT water level never occurs, the operator is credited for manually actuating ADS. As with automatic ADS, the manual ADS success paths can be subdivided into full ADS (Section A3.3.1) and partial ADS (Section A3.3.2).

There are also success paths that model the combination of full ADS actuation and RNS injection. These success paths are listed on Table A3.1-2, but are not further discussed in Section A3.2 or Section A3.3. This is because if full ADS actuation has occurred and it has been shown that the RCS pressure is low enough to allow IRWST gravity injection, then certainly the RCS depressurization is sufficient to allow RNS pumped injection. Or, if RNS injection is shown to lead to core cooling with minimal venting through partial ADS, the greater venting with full ADS is sure to keep the RCS sufficiently depressurized for RNS injection.

A.3.2 Automatic ADS Actuation

The ADS actuation logic for AP1000 is the same as it was for AP600, with the actuation signal coming from the low CMT level (ADS stages 1, 2 and 3) and the low-low CMT level (ADS stage 4). The PRA success paths that automatically actuate ADS based on the draining of a CMT are discussed in the following sections, with the IRWST gravity injection success paths discussed in Section A3.2.1, and the RNS pumped injection success paths discussed in Section A3.2.2. These sections show successful core cooling within the time frame of establishing IRWST gravity injection or RNS pumped injection. Long term cooling is addressed in Section A3.5.

A.3.2.1 Automatic ADS Leading to IRWST Gravity Injection

This section discusses the basis for successful core cooling for success paths including successful CMT actuation, successful ADS actuation, and successful IRWST gravity injection. The ADS actuation for these success paths is termed "full depressurization" since gravity injection does not start until the ΔP between the RCS and containment has been reduced to approximately 15 psia. Full depressurization requires 3 stage 4 ADS lines to open. The success paths addressed within this section are listed in Table A3.2-1, which summarizes the success criteria that make up the event paths.

The equipment being credited in these success paths is the same as modeled for AP600 in the first four MAAP4 and NOTRUMP benchmarking cases in Reference A-3, which address break sizes from 0.5 inches to 8.75 inches. Specifically, the limiting equipment assumptions are:

- One CMT injects water into the RCS.
- No accumulators inject water into the RCS. (There is no top event for accumulators applied to these success paths, and thus accumulator success/failure is not known. But the accumulators are not credited because they have not been defined as part of the success criteria for these paths.)
- No stage 1, stage 2 or stage 3 ADS valves are credited. The exception to this is breaks with a diameter less than 2 inches credit 1 stage 3 ADS valve to reduce the RCS below the stage 4 interlock pressure if the PRHR HX is not available.
- Three stage 4 ADS valves are credited to automatically open, based on the low-low CMT level signal, plus the applicable time delay.

- The PRHR is not credited. (PRHR is generally not a part of the success criteria for these success paths. However, in some of the high pressure events the PRHR is a part of the success criteria, but only to the extent that it would fulfill the same purpose as stage 2 or stage 3 ADS to depressurize below the stage 4 interlock pressure.)

- One valve in one DVI line opens to allow a path for IRWST gravity injection.

- Containment isolation failure is assumed.

With the same equipment assumptions for the AP1000 success criteria for automatic, full ADS success paths compared to the AP600 analyses in Reference A-3, the basis for successful core cooling lies in the similarity of the AP1000 plant response compared to the AP600 plant response. Although the AP1000 plant has a higher core power, compensating design changes were made to result in similar plant response. The design changes, compared to AP600, which apply to these success paths are the larger CMT, the larger stage 4 ADS valves, and the larger IRWST lines. In addition, the CMT injection characteristics have been changed, such that there is not only more water mass, but it is delivered at a faster rate.

Figure A3.2-1 shows the minimum core mixture level as a function of break size for AP1000 when there is automatic ADS leading to IRWST gravity injection. The plot differentiates the minimum mixture level that occurs during the initial blowdown and period of CMT injection compared to the minimum mixture level after ADS actuation. As was observed for AP600, the CMT injection is adequate to provide core cooling for the high pressure events and for LOCAs up to 8.75 inches prior to ADS actuation. The minimum core mixture level occurs at the largest breaks during this time period. As shown in Reference A-3, there could be a slight period of core uncovery as CMT injection is established for these largest breaks, but as shown later within this section, the AP1000 plant response is similar or less limiting than the AP600 plant response. After ADS actuation, core uncovery can occur due to the coolant inventory lost during the ADS blowdown, until IRWST injection is established. The AP1000 plant response is again similar to the AP600 plant response, with the most limiting core uncovery after ADS actuation occurring for the smaller break sizes.

The AP1000 plant response is directly compared to the AP600 plant response for four representative cases. The following cases were selected for the automatic ADS, full depressurization category:

- 0.5 inch hot leg break
- 2.0 inch hot leg break
- 5.0 inch hot leg break
- 8.75 inch hot leg break

These cases represent a high pressure scenario, as well as samples of the plant response across the MLOCA break spectrum. They are selected to be close to the limiting break sizes in regards to the core uncovery. In addition, the AP600 plant response to these cases was documented in Reference A-3, based on both MAAP4 and NOTRUMP analyses.

For each case, plots are provided of six key system parameters, showing the AP1000 response compared to the AP600 plant response.

1. RCS pressure
2. CMT water mass
3. Integrated break water and steam release
4. Integrated stage 4 ADS vapor release
5. Integrated water injection from IRWST gravity draining
6. Core mixture level

The plots are contained in Figures A3.2-2 to A3.2-7 for the 0.5 inch case, Figures A3.2-8 to A3.2-13 for the 2.0 inch case, Figures A3.2-14 to A3.2-19 for the 5.0 inch case, and Figures A3.2-20 to A3.2-25 for the 8.75 inch case.

In all cases, the depth and duration of core uncovery for AP1000 is the same or less limiting than seen for AP600. The faster CMT injection for AP1000 is a benefit during the first part of the transient, and the larger stage 4 ADS and IRWST lines cause less core uncovery later in the transient. Because the AP1000 plant response is similar or better than that shown to be successful core cooling for AP600, the success paths and the associated success criteria identified in Table A3.2-1 remain valid for AP1000.

A3.2.2 Automatic ADS Leading to RNS Injection

This section discusses the basis for successful core cooling for success paths including successful CMT actuation, successful partial ADS actuation, and successful RNS pumped injection. The ADS actuation for these success paths is termed "partial depressurization" because stage 4 ADS valves are not required and only two stage 2 or stage 3 ADS or one stage 4 valves are credited. RNS injection can start when the RCS pressure is below 175 psia, and thus the RCS does not have to be fully depressurized. The success paths addressed within this section are listed in Table A3.2-2, which summarizes the success criteria that make up the event paths.

There are two initiating events that do not credit RNS injection – DVI line break and station blackout. In the DVI line break event, it is assumed that the break occurs in the line through which RNS water would be delivered to the RCS. In the station blackout event, the loss of offsite power with the loss of diesel generators incapacitates the RNS pumps. However, there is an automatic ADS success path leading to RNS injection for all the other initiating events for which there was a full depressurization success path (discussed in Section A3.2.1).

The limiting or minimum set of equipment being credited in these success paths is the same as the full depressurization cases except the ADS valves and method of injection. Specifically, the limiting equipment assumptions are:

- One CMT injects water into the RCS.
- No accumulators inject water into the RCS. (There is no top event for accumulators applied to these success paths, and thus accumulator success/failure is not known. But

the accumulators are not credited because they have not been defined as part of the success criteria for these paths.)

- Two stage 3 ADS valves are credited to automatically open, based on the low CMT level signal, plus the applicable time delay. (The success criterion is for two valves of either stage 2 or stage 3 to open. Stage 3 valves are selected as slightly more limiting because of the longer delay time until they open.)
- No stage 1, stage 2, or stage 4 ADS valves are credited.
- The PRHR is not credited. (PRHR is a part of the success criteria for some of the high pressure initiating events. However, even for smaller breaks, it is not a necessary element to achieving successful core cooling in these success paths, and is conservatively ignored.)
- One RNS pump provides injection from the cask loading pit outside containment.

The success criteria for these success paths have been kept the same as AP600 with the exception that the number of stage 2 or stage 3 ADS valves has been increased from 1 to 2. In addition, the merging of the NLOCA and MLOCA event trees from AP600 to the single MLOCA event tree for AP1000 results in a change in the 6 inch to 9 inch break range. For AP600, the success paths for this break size did not require ADS valves to achieve RNS injection, while for AP1000 there is no distinction made for breaks that would be large enough to depressurize to RNS injection without ADS actuation. Also, the source water for RNS injection has been moved to outside the containment, but the same system delivery curve (flowrate versus RCS pressure) is modeled. As noted in Section A3.2.1, the CMT has more water mass with faster flow characteristics than AP600, but nothing has changed in the CMT success criterion.

Figure A3.2-26 shows the minimum core mixture level as a function of break size for AP1000 when there is automatic partial ADS leading to RNS pumped injection. The plot differentiates the minimum mixture level that occurs during the initial blowdown and period of CMT injection compared to the minimum mixture level after ADS actuation. As seen in Section A3.2.1, the CMT injection is adequate to provide core cooling for the high pressure events and for LOCAs up to 8.75 inches prior to ADS actuation. After ADS actuation, core uncovery can occur due to the coolant inventory lost during the ADS blowdown, until RNS injection is established. However, the opening of two stage 3 ADS valves allows earlier pumped RNS injection than the IRWST gravity injection that occurs when three stage 4 ADS valves are opened. Figures A3.2-27 to A3.2-34 contain plots of the RNS injection and core mixture level transient for the 0.5 inch, 2.0 inch, 5.0 inch and 8.75 inch breaks. The figures compare the RNS injection/partial depressurization cases to the IRWST injection/full depressurization cases from Section A3.2.1.

The automatic ADS success paths leading to RNS injection, as listed in Table A3.2-2, have been shown to result in successful core cooling through the time period in which RNS injection is established.

A3.3 Manual ADS Actuation

The ADS actuation logic for AP1000 is the same as it was for AP600, with the actuation signal coming from the low CMT level (ADS stages 1, 2 and 3) and the low-low CMT level (ADS stage 4). However, if CMT valves fail to open and neither CMT drains, the automatic ADS signals are never generated. The PRA event trees model the possible accident progression when neither CMT injects and the only method of ADS actuation is to credit operator action. Success paths that credit manual ADS actuation are discussed in the following sections, with the IRWST gravity injection success paths discussed in Section A3.3.1, and the RNS pumped injection success paths discussed in Section A3.3.2. These sections show successful core cooling within the time frame of establishing IRWST gravity injection or RNS pumped injection. Long term cooling is addressed in Section A3.5.

A3.3.1 Manual ADS Leading to IRWST Gravity Injection

This section discusses the basis for successful core cooling for success paths including successful accumulator injection, successful operator manual actuation of stage 4 ADS, and successful IRWST gravity injection. The ADS actuation for these success paths is termed "full depressurization" since gravity injection does not start until the ΔP between the RCS and containment has been reduced to approximately 15 psia. Full depressurization requires 3 stage 4 ADS lines to open. The success paths addressed within this section are listed in Table A3.3-1, which summarizes the success criteria that make up the event paths.

The equipment being credited in these success paths is the same as modeled for AP600 in three of the MAAP4 and NOTRUMP benchmarking cases in Reference A-3, which analyze limiting break sizes from 3.5 inches to 8.75 inches. Specifically, the limiting equipment assumptions are:

- Both CMTs fail to inject water into the RCS
- PRHR HX operation
- One accumulator injects water into the RCS
- No stage 1, stage 2, or stage 3 ADS valves are credited
- Three stage 4 ADS valves are credited to manually open at 20 minutes after the failed CMT actuation signal
- One valve in the DVI line opens to allow a path for IRWST gravity injection
- Containment isolation failure is assumed

Note that credit for the PRHR HX is required for some of these breaks, mainly between approximately 3 inches and 4 inches. In this break range, notable RCS coolant is lost in the first 20 minutes, but the RCS does not depressurize enough to allow substantial accumulator injection when the PRHR is not credited. With failed CMTs and the RCS pressure too high for accumulator injection, there is no source of makeup water until the operators manually actuate the ADS. While successful core cooling was shown for AP600 with this set of conditions until operator action at 20 minutes, the depth and duration of core uncovery would increase for AP1000 if the same timing and equipment assumptions were maintained. Therefore, the PRHR has been added as a success criterion for the manual ADS success paths for MLOCAs and associated event trees with similar-sized breaks. The PRHR helps to reduce RCS pressure, which both reduces the break flowrate and allows accumulator injection to begin before ADS actuation for smaller break sizes. Note that in the following cases analyzed with the PRHR HX operation, the PRHR HX is conservatively assumed to stop functioning when the accumulator empties and discharges nitrogen into the RCS.

Figure A3.3-1 shows the minimum core mixture level as a function of break size for AP1000 when there is manual ADS leading to IRWST gravity injection. The plot differentiates the minimum mixture level that occurs during the initial blowdown compared to the minimum mixture level around the time of ADS actuation. During the initial blowdown, the smallest breaks do not lose enough coolant inventory to challenge core uncovery, and larger breaks (approaching 9 inches) get accumulator injection to prevent core uncovery. But the middle break sizes, between approximately 3 and 6 inches, may experience a decrease in the core mixture level as there is a trade-off between the break size and the amount of accumulator injection that can occur.

After ADS actuation, all break sizes show that there may be some core uncovery before adequate IRWST gravity injection is established. With manual actuation causing similar timing of ADS regardless of the break size, there is less variation in the plant response after ADS than occurs in the automatic ADS cases. For smaller break sizes, the ADS stage 4 area is the major path of venting and depressurizing the RCS. There is some improvement in the ability to achieve IRWST gravity injection as the break size increases. However, as the break size increases to the upper end of the MLOCA break spectrum (up to 9 inch diameter), the trend changes and MAAP4 predicts the deepest core uncovery. This is because the accumulator empties prior to ADS actuation, and a period of no makeup inventory occurs with a relatively high break flow rate. As noted in Reference A-3, the homogeneous core void fraction in MAAP4 tends to underpredict the amount of water from the accumulator that can be stored within the reactor vessel. As the accumulator injects, this results in a loss of more coolant out the break and a prediction of earlier core uncovery than a more detailed code like NOTRUMP.

It is also noteworthy that the minimum vessel mixture level in Figure A3.3-1 has been determined with analyses using an atmospheric containment back pressure. While the analyses are to address a failure in the containment isolation system, the use of an atmospheric containment pressure pessimistically delays the start of the IRWST gravity injection. Even if there were a failure in the containment isolation system, some containment pressurization would be expected, especially as stage 4 ADS is actuated. The higher containment pressure would make it easier to vent the decay heat, and IRWST gravity injection would be established earlier, maintaining a higher mixture level within the core.

The AP1000 plant response is directly compared to the AP600 plant response for three representative cases. The following cases were selected for the manual ADS, full depressurization category:

- 3.5 inch hot leg break
- 6.0 inch hot leg break
- 8.75 inch hot leg break

These cases represent samples of the plant response across the MLOCA break spectrum. They are selected to be close to the limiting break sizes in regards to the core uncovery. In addition, the AP600 plant response to these cases was documented in Reference A-3, based on both MAAP4 and NOTRUMP analyses.

For each case, plots are provided of the following six key system parameters, showing the AP1000 response compared to the AP600 plant response:

1. RCS pressure
2. Accumulator water mass
3. Integrated break water and steam release
4. Integrated stage 4 ADS vapor release
5. Integrated water injection from IRWST gravity draining
6. Core mixture level

The plots are contained in Figures A3.3-2 to A3.3-7 for the 3.5 inch case, Figures A3.3-8 to A3.3-13 for the 6.0 inch case, and Figures A3.3-14 to A3.3-19 for the 8.75 inch case.

In the 3.5 inch and 6.0 inch cases, the depth and duration of core uncovery for AP1000 is the same or less limiting than seen for AP600. Crediting the PRHR has a beneficial impact during the first part of the transient, and the larger stage 4 ADS and IRWST lines cause less core uncovery later in the transient.

In the 8.75 inch case, however, the core uncovery predicted by MAAP4 is greater for AP1000 than AP600. This is partially due to the accumulator for AP1000 being the same size as for AP600, and yet the AP1000 reactor vessel is bigger and there is more coolant mass lost out the break for AP1000. The accumulator also depletes faster for AP1000 than AP600, perhaps aided by the PRHR heat removal. As noted above, the MAAP4 prediction of core uncovery for this case is also known to err on the conservative side. Nevertheless, the depth and duration of core uncovery, as predicted by MAAP4, is within the range that would lead to peak cladding temperatures of less than 2200°F. It is also noteworthy that any larger break would fit into the large LOCA initiating event, which not only requires at least one CMT in addition to the accumulator success criterion, but does not credit any manual ADS success path.

The success paths and the associated success criteria identified in Table A3.3-1 have been shown to result in successful core cooling for AP1000.

A3.3.2 Manual ADS Leading to RNS Injection

This section discusses the basis for successful core cooling for success paths including successful accumulator injection, successful operator actuation of stage 2 or stage 3 ADS, and successful RNS pumped injection. The ADS actuation for these success paths is termed "partial depressurization" since RNS injection can start when the RCS pressure is below 175 psia, and thus the RCS does not have to be fully depressurized. The success paths addressed within this section are listed in Table A3.3-2, which summarizes the success criteria that make up the event paths.

There are two initiating events that do not credit RNS injection – DVI line break and station blackout. In the DVI line break event, it is assumed that the break occurs in the line through which RNS water would be delivered to the RCS. In the station blackout event, the loss of offsite power with the loss of diesel generators incapacitates the RNS pumps. However, there is an manual ADS success path leading to RNS injection for all the other initiating events for which there was a full depressurization success path (discussed in Section 3.2.1).

The limiting or minimum set of equipment being credited in these success paths is the same as the full depressurization cases except the number and type of ADS valves and method of injection. Specifically, the limiting equipment assumptions are:

- Both CMTs fail to inject water into the RCS
- PRHR HX operates
- One accumulator injects water into the RCS
- Two stage 3 ADS valves are credited to manually open at 20 minutes after the failed CMT actuation signal. (The success criterion is for two valves of either stage 2 or stage 3 to open. Since these valves are the same size and there is no timing difference when operator actuation is modeled, it does not matter whether the opened valves are stage 2 or stage 3 ADS.)
- No stage 1, stage 2 or stage 4 ADS valves are credited
- One RNS pump provides injection from the RWST outside containment

The differences in the AP1000 success criteria compared to the AP600 success criteria is the number of stage 2 or stage 3 ADS valves has been increased from 1 to 2, and the PRHR is credited for AP1000 for MLOCA, CMT line break, and DVI line break. The addition of the PRHR for manual ADS actuation cases was discussed in Section A3.3.1. In addition, the merging of the NLOCA and MLOCA event trees from AP600 to the single MLOCA event tree for AP1000 results in a change in the 6 inch to 9 inch range. For AP600, the success paths for this break size did not require ADS to achieve RNS injection, while for AP1000 there is no distinction made for breaks that would be large enough to depressurize to RNS injection without ADS actuation. Also, the source water for RNS injection has been moved to outside the containment, but the same system delivery curve (flowrate versus RCS pressure) is modeled.

Figure A3.3-20 shows the minimum core mixture level as a function of break size for AP1000 when there is manual ADS leading to RNS pumped injection. There is no core uncovery at any break size for these set of success criteria. The general plant behavior during the initial blowdown and accumulator injection period is the same as shown in Section A3.3.1. The plant response after ADS is opened is similar to that shown in Section 3.2.2, demonstrating that 2 stage 3 ADS valves are sufficient to depressurize the RCS to allow adequate RNS injection.

Figure A3.3-21 shows an overview of the timing of accumulator injection, ADS actuation and RNS injection that occurs as the break size changes. Note that at the upper end of the MLOCA break spectrum, RNS injection is able to start before ADS valves are opened.

The success paths and the associated success criteria identified in Table A3.3-2 have been shown to result in successful core cooling for AP1000.

A3.4 Large LOCA Success Criteria

There are two large break LOCA event trees used in the AP1000 PRA. One includes breaks of the hot leg (HL) or cold leg (CL) pipes, up to and including the double ended rupture of the main loop lines. The other includes the spurious opening of the ADS valves, up to and including opening of all 4 ADS stage 4 valves at the same time.

Large Break LOCA (LLOCA)

This LOCA is defined as a break sufficiently large such that injection from both accumulators is required. Operation of ADS valves is not required in order to depressurize the RCS to the RNS injection pressure. Operation of ADS valves is required in order to depressurize the RCS to allow gravity injection from the IRWST and containment recirculation. The corresponding break size is a break with an equivalent inside diameter of approximately 9 inches or larger.

The DCD Chapter 15 analysis covers this event since it also assumes operation of both accumulators. As a result, special PRA success criteria analysis is not required.

Spurious ADS LOCA (SPADS)

The opening of all four ADS stage 4 paths bounds this LOCA. Although this LOCA size is within the LLOCA size, the AP1000 success criteria is one accumulator because of the less severe plant response to HL LOCAs as compared with large CL LOCAs. Otherwise the mitigating system requirements are the same as for a LLOCA.

The upper bound of this event is the spurious opening of all 4 ADS-4 valves at the same time. Since the AP1000 PRA success criteria for this event is 1 accumulator, the design basis DCD analysis does not bound the PRA case. The analysis of this accident shown in Section A.5 provides a conservative evaluation of the response of AP1000 to this accident. It shows that the PCT is less than 1100°F even with uncertainties added.

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. If would be even less for the plant design life of 60 years.

The following ATWS T/H analysis has three significant conservatisms. One is that the reactivity feedback during an ATWS transient as the reactor temperature increases is underestimated. The second is that the SG heat transfer remains very high until the SG dries out and then it suddenly drops to zero. Studies done for other plants have indicated that these two effects would significantly reduce the peak RCS pressure such that ATWS would be acceptable during most if not all of the first core cycle and the AP1000 UET would be essentially zero. The third is that the pressurizer safety valve capacity was assumed to be just equal to the system requirement for the valve; the valve provided will have a greater rated capacity.

A4.2 ATWS Analysis

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. As was 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.

In these analysis, 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 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.

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 A5.1.3. 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.

A5.1.4 Screening Criteria for Dominant Sequences

The screening criteria for identifying a success path as "dominant" are as follows:

> A success path is considered as dominant if either its postulated CDF or LRF frequency is 1 percent or more of the base CDF or LRF frequency. This postulation is done by tentatively assuming that the path leads to core damage for evaluation purposes.

The base AP1000 CDF and LRF frequencies are 2.41E-07/year and 1.95E-08/year, respectively.

In addition, the total frequency of the residue is required to be small with respect to uncertainty analysis. Namely, even when all the residual sequences are assumed to go to CDF (which is overly conservative), the total CDF should not increase by more than a factor of two. The same applies to LRF.

Note that the expanded event trees do not include available success paths using nonsafety features. For example, if three ADS Stage 4 valves are not available for an MLOCA, then the normal residual heat removal system (RNS) pumps could still provide adequate injection. However, in the expanded event trees, if three of four ADS Stage four valves are not available, then the sequence is considered a possible core melt sequence. This is the same approach done for AP600.

A5.2 Expanded Success Paths

The event tree success paths are expanded the same as in the T/H uncertainty analysis made for AP600. Since the full ADS success criteria is three out of four stage 4 lines being available, many of the success paths that showed up in AP600 no longer show up in AP1000 expanded event trees (namely, the two of four ADS Stage 4 paths being successful); this reduces the paths subject to potential T/H uncertainty analysis.

The success path end states are defined in Table A5.1-1 (similar to those in Reference A-4).

A5.2.1 Large LOCA Event Tree

The LLOCA event tree success criteria for the AP1000 PRA has been developed and provided in the AP1000 PRA. The core damage event tree for the LLOCA event is given in Chapter 4. Based on these references, an expanded LLOCA event tree has been developed (along the same lines as in Reference A-4) and is given in Figure A5.1-1. This expanded event tree provides the various combinations of success paths in detail. Note that the only UC end states that may be candidates for T/H uncertainty analysis are those when containment isolation is assumed to fail. Otherwise, the DCD Chapter 15 analysis covers the LLOCA event with the assumption that both accumulators are operable.

A5.2.2 Spurious Automatic Depressurization System Event Tree

The Spurious ADS category is a variation of LLOCA, where the size and location of the break is known due to the nature of the event. Because the ADS stage 4 valves are connected to the hot legs, less accumulator injection is required and only one of two accumulators is required as opposed to two of two for a large cold-leg LOCA. Otherwise, the success criteria and event tree logic are the same as that of the LLOCA core damage event tree. An expanded event tree has been developed for the Spurious ADS category, as given in Figure A5.1-2.

A5.2.3 Medium LOCA Event Tree

The MLOCA event break range for the AP1000 includes both the MLOCA and NLOCA event tree ranges for the AP600. The additional credit taken for NLOCA in the AP600 is not taken for the AP1000. This helps remove some of the T/H uncertainty.

The MLOCA expanded event tree is shown in Figure A5.1-3.

A5.2.4 Safety Injection Line Break Event Tree

The SI-LB event tree success criteria for the AP1000 PRA has been developed and provided in the AP1000 PRA. The core damage event tree for the SI-LB event is given in Chapter 4. Based on these references, an expanded SI-LB event tree has been developed (along the same lines as in Reference A-4) and is given in Figure A5.1-4. This expanded event tree provides the various combinations of success paths in detail.

A5.2.5 Core Makeup Tank Line Break Event Tree

The CMTLB event tree is a special case of the MLOCA event tree, with the constraint that at most one CMT is available due to the nature of the initiating event.

An expanded event tree has been developed for the CMTLB category, as given in Figure A5.1-5.

A5.2.6 Success Paths with Normal Residual Heat Removal

Some success paths in the MLOCA and CMTLB event trees credit use of normal residual heat removal (RNS). Since the RNS is an "active" system, it is not considered to be a significant contributor to T/H uncertainty. Thus, the success paths containing RNS in MLOCA and CMTLB trees are not further expanded, and are not included in the selection of the dominant success paths for further evaluation.

Moreover, the system importance of the RNS was already calculated to be low for the AP1000 PRA internal events at power (CDF increases by a factor of 1.7 if the RNS is assumed to be inoperable across the board in all events). This system importance is much smaller than those calculated for passive systems (Table 50-12 of the AP1000 PRA).

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 A5.1.3 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 of the low PCTs calculated for the case (< 1100°F with uncertainty).

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 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

low; the containment pressure is higher early in the accident progression, especially after Stage 4 ADS actuation.

The applicability of NOTRUMP to AP1000 design basis accidents and PRA scenarios has already been presented in References A-4 and A-14.

A5.5.1.2 LOCTA Cladding Heat-Up Methodology

When a T/H uncertainty case, results in noticeable core uncovery, a cladding heatup analysis is performed. The cladding heatup analysis is used to determine if adequate core cooling is maintained for the T/H uncertainty scenario.

The Westinghouse small-break cladding heatup code (LOCTA) of Reference A-16, as modified by Reference A-17, is used to determine the peak cladding temperature of the lead rod. The cladding heat-up code (LOCTA) applies to the AP1000 design because:

1. The AP1000 uses the 17x17 XL Robust fuel design already in use in conventional Westinghouse-designed PWRs. This fuel type lies within the assumptions and models employed in the code.

2. The low pressures seen in the AP1000 small-break transients subsequent to ADS actuation are within the limits of the small-break heat-up code, since this code is a version of the Westinghouse large-break heat-up code, which must operate at low pressures.

3. The AP1000 uses a 14-foot core design and has peaking factors similar to current Westinghouse-designed operating reactors. Thus, the power shape used in the AP1000 cladding heatup calculation was taken from data for current core designs.

Additionally, the cladding heatup analysis assumes a total core peaking factor (F_Q) of 2.60 and an enthalpy rise peaking factor ($F_{\Delta H}$) of 1.65.

A5.5.1.3 NOTRUMP/LOCTA Results

The small LOCA cases outlined in Table A5.1-6 are LOCA scenarios with breaks either in the RCS hot leg, CMT balance line, or the DVI line piping. The response of the AP1000 plant to these events are presented in the following subsections.

A5.5.1.3.1 Case A Results

Case A is a 3.0-inch break in the RCS hot leg. This break size is the maximum that will keep the RCS pressure at or above the accumulator pressure (700 psia) at the time that manual ADS-4 actuation is assumed (1200 seconds after the safety injection signal). Neither of the 2 CMTs is assumed to operate and, therefore, operator action to actuate the ADS must be assumed. Additional assumptions are:

- Credit for PRHR HX operation

- Credit for only 1 of 2 accumulators
- ADS stages 1, 2, and 3 ADS fail to open
- Credit for 4 out of 4 ADS stage 4 at 20 minutes (1200 seconds) after safety injection (SI) signal
- Only 1 of 2 IRWST lines is assumed available for injection. Further, failure of 1 of the 2 parallel paths in the available IRWST line is assumed
- Containment pressure is assumed to be 14.7 psia. Containment pressures greater than 14.7 psia have been shown to improve the performance of the passive safety systems. Consequently, the containment pressure for this case is conservatively assumed to be 14.7 psia.

Figures A5.2-1 through A5.2-13 provide plots of the plant response and Table A5.2-1 provides the sequence of key events. Figures A5.2-3 and A5.2-4 show the liquid and steam break flow rates that lead to depressurization of the RCS, as seen in Figure A5.2-1, and draining of the RCS pressurizer (Figure A5.2-2). The 3.0-inch break size was selected as the largest break size that together with the PRHR HX would not result in significant accumulator injection before the operator opens the ADS stage 4 valves at 20 minutes. Figure A5.2-1 shows the RCS pressure to be slightly below the accumulator cut-in pressure of 715 psia at 20 minutes. At 20 minutes, the operator opens all 4 ADS stage 4 valves, which results in rapid depressurization down to less than 50 psia. The accumulator injects as a result of the depressurization, refilling the RCS downcomer, and recovering the core. The single accumulator runs dry at about 1430 seconds and the IRWST begins to inject. Core uncovery occurs before operator action to open the ADS stage 4 valves, followed by a rapid recovery of the core due to injection of the single accumulator. A minimum RCS mass of 60,000 lbm occurs shortly after 1200 seconds (Figure A5.2-12), the time of maximum core uncovery.

A cladding heatup calculation for case UC1 (Figure A5.2-13) shows a peak cladding temperature of 719°F at 13.75 feet on the fuel rod occurring at 1238 seconds. These results are well below the 2200°F acceptance criterion.

A5.5.1.3.2 Case B Results

Case B is a double-ended rupture of an 8.0-inch CMT balance line (inside diameter of 6.8 inches). This break is very much like a break in the RCS cold leg. Both CMTs are assumed to fail. In addition, the break is assumed to be in a location that prevents the faulted CMT from draining. Therefore, operation action to actuate the ADS must be assumed.

- Credit for PRHR HX operation
- Credit for 2 out of 2 accumulators
- ADS stages 1, 2, and 3 fail to open
- Credit for 4 out of 4 ADS stage 4 at 20 minutes (1200 seconds)

- Only 1 of 2 IRWST lines is assumed to inject. Further, failure of 1 of the 2 parallel paths in the IRWST line to open is assumed

- Credit for containment isolation; containment pressure assumed to be 25 psia, which was calculated for the DEDVI break. Since the DE CMT balance line is a larger break, the 25 psia containment pressure is conservative for this case.

Figures A5.2-14 through A5.2-25 provide plots of the plant response and Table A5.2-2 provides the sequence of key events. Figures A5.2-16 and A5.2-17 show the liquid and steam break flow rates that lead to depressurization of the RCS, as seen in Figure A5.2-14, and draining of the RCS pressurizer (Figure A5.2-15). Due to the large size of the break and lack of CMT injection, the RCS rapidly depressurizes and accumulator injection begins at around 290 seconds. Both accumulators continue to inject until around 1350 seconds, providing adequate injection to keep the core covered. At 20 minutes, the operator opens all 4 ADS stage 4 valves, which results in a further depressurization down to less than 50 psi. The depressurization brought on by the opening of ADS stage 4 is sufficient to allow for IRWST injection, which begins at 1450 seconds (250 seconds after opening ADS stage 4). The IRWST injection rate is sufficient to prevent core uncovery, stabilizing at about 150 lbm/sec, which matches the losses out of the break and ADS. Since core uncovery does not occur for case UC2B, the clad does not experience a heat-up, and a clad heat-up calculation is not performed.

A5.5.1.3.3 Case C Results

Case C is an double-ended rupture of the DVI line piping. On the vessel side, the break is limited to 4 inches in diameter by an orifice. On the passive injection side, the break is limited by the CMT discharge orifice. Additional assumptions are:

- The CMT on the intact loop provides injection to the RCS. The CMT isolation valve on the faulted loop is assumed to fail to open.

- Both accumulators fail to inject.

- ADS stages 1, 2, and 3 fail to open.

- Credit for 3 of 4 ADS stage 4, automatically actuated due to draining of the intact CMT.

- Only 1 of 2 IRWST lines is assumed to inject. Further, failure of 1 of the 2 parallel paths in an IRWST line to open is assumed.

- No credit for containment isolation; containment pressure assumed to be 14.7 psia.

Figures A5.2-26 through A5.2-36 provide plots of the plant response and Table A5.2-3 provides the sequence of key events. Figures A5.2-28 and A5.2-29 show the liquid and steam break flow rates on the vessel side of the broken DVI piping, which leads to RCS depressurization as seen in Figure A5.2-26, and draining of the RCS pressurizer (Figure A5.2-27). The intact CMT begins to recirculate at about 40 seconds and drain down starts at 280 seconds. The intact CMT drains, resulting in ADS stage 4 actuation at

1380 seconds. The actuation of ADS stage 4 results in a depressurization down to less than 50 psia. The depressurization brought on by the opening of ADS stage 4 is sufficient to allow for IRWST injection, which begins at 1960 seconds (580 seconds after ADS stage 4 opens). The IRWST injection rate is sufficient to recover the core, exceeding losses through the break and ADS stage 4 at 2890 seconds.

Figure A5.2-37 shows that the cladding heatup as calculated by LOCTA is comfortably below the 2200°F acceptance criteria. Note that this case is a conservative case that bounds two different risk significant cases. The actual risk significant cases will have less core uncovery and lower PCTs.

A5.5.2 WCOBRA/TRAC Analysis of Large-Break LOCA

Westinghouse applies the WCOBRA/TRAC computer code to perform AP1000 best-estimate large-break LOCA analyses in compliance with 10 CFR 50 (in the DCD). The methodology used for the AP1000 analysis is documented in References A-22 and A-23.

The acceptability of WCOBRA/TRAC computer code and methodology approved for AP600 large-break LOCA analyses for the AP1000 application is documented in Reference A-24.

A simplification of this methodology was approved for the AP600 in Reference A-25. The parameters important to the initial conditions and power distribution uncertainty components are set to bounding values established by sensitivity studies. The model uncertainty component is quantified in the same way as for three- and four-loop plants, with the other parameters set to those bounding values. The code uncertainty estimate based on direct comparisons with data, the uncertainty in the experimental data itself, is also considered in the overall uncertainty estimate. A discussion of the large-break LOCA uncertainty methodology is given in Reference A-23.

A5.5.2.1 Case D Results

Thermal/hydraulic (T/H) uncertainty Case D is a large cold-leg loss-of-coolant accident (LOCA) that is the same as the CD = 1.0 double-ended cold-leg guillotine (DECLG) break reference case presented in the base case of the AP1000 Design Control Document (DCD) Section 15.6, 95th percentile peak cladding temperature (PCT) determination, with the following exceptions:

- Containment isolation has failed, so that the containment is at atmospheric pressure.
- Offsite power is available until the reactor coolant pumps are tripped when the core makeup tanks (CMTs) are actuated several seconds into the event.

Note that if offsite power is assumed to be unavailable, this sequence would not be risk-important. This case is analyzed using the same computer code (WCOBRA/TRAC) with the same assumptions as the DECLG reference case from the DCD with the exceptions discussed above.

The results are discussed in the following paragraphs.

A WCOBRA/TRAC analysis has been performed of this DECLG LOCA with both accumulators available. Attached are plots of the analysis results from the WCOBRA/TRAC analysis of this case. Figure A5.2-38 provides the reactor vessel pressure transient during Case D. Figures A5.2-39, A5.2-40, and A5.2-41 show the flow rates at the top of the core hot assembly, the fuel assemblies beneath the upper core plate open holes, and the fuel assemblies beneath the guide tube locations, respectively; in each of these figures, the solid line is the vapor flow rate, and the dashed lines are the continuous liquid (FLM) and entrained liquid (FEM) flow rates. Figure A5.2-42 provides the core collapsed liquid level during Case D to indicate the voiding during blowdown and the subsequent reflooding of the core. Figure A5.2-43 shows the cladding temperature (PCT) for Case D at the peak elevation for any time during the transient. The calculated PCT is 1628°F.

Approximately 5 seconds into the transient, a brief period of positive liquid flow occurs throughout the core. The flow, while short-lived, provides adequate blowdown cooling to terminate the cladding temperature excursion temporarily. This results in a relatively low predicted PCT for the blowdown phase. Eventually, the reflood phase calculated PCT is also lower than in the DCD Chapter 15 large-break LOCA analysis case, which presumes a loss of offsite power at the time of the break.

To estimate the 95th percentile PCT value for Case D, the difference between the DCD large-break LOCA analysis reference case calculated PCT and the licensing basis result with uncertainty considered is applied. The difference (2124-1896) = 228°F, so the estimated Case D PCT at the 95th percentile is (1628+228) = 1856°F. The Case D scenario result exhibits large margin to the regulatory limit of 2200°F.

A5.5.2.2 Case E Results

Case E is a spurious opening of all four ADS stage 4 valves. The scenario discussed in Section A5.1 has been analyzed on a best-estimate basis using the WCOBRA/TRAC computer code and the AP1000 input from the DCD large-break LOCA analysis. The peak cladding temperature (PCT) calculated by WCOBRA/TRAC is 833°F. This result is less limiting than the corresponding DECLG break reference case result with both accumulators available that is presented in the AP1000 DCD and is the base case of the AP1000 DCD Section 15.6 95th percentile PCT determination.

Figure A5.2-44 presents the PCT transient for the hot rod of the AP1000 core. Because the flow to the break location is in the normal operation flow direction upward through the fuel, there is no flow reversal immediately following the break to cause DNB to occur in the core. As a result, and due to the strong positive liquid flow through the core, there is no blowdown cladding heatup. Figure A5.2-45 shows the continuous liquid phase flow rates at the top (dashed line) and bottom (solid line) of the core until the cladding heatup has begun. Core pressure (Figure A5.2-46) is reduced to about 300 psia at the time the cladding temperature excursion begins.

Depletion of the reactor vessel mass inventory due to the flow through the open ADS-4 valves eventually leads to cladding heatup due to the lack of liquid flow through the core. Figures A5.2-47 and A5.2-48 present core and downcomer liquid levels, respectively, and show the loss in mass inventory that occurs through the time that the cladding temperature excursion begins as well as the subsequent increase in mass. The diminished liquid available leads to the low liquid flow rates through the core observed in Figure A5.2-45. Flow from the one accumulator assumed to be operable (Figure A5.2-49) is what causes the level increase observed in Figures A5.2-47 and A5.2-48. The accumulator initial conditions of water level, gas pressure, and discharge line resistance used in this calculation are the conservative values used in the AP1000 DCD Chapter 15 large-break LOCA analysis.

The addition of an appropriate PCT uncertainty to the WCOBRA/TRAC best-estimate PCT result will conservatively address thermal/hydraulic analysis uncertainties. The AP1000 DCD subsection 15.6, 95th percentile PCT value of 2124°F is 228°F higher than the WCOBRA/TRAC reference case result from the DCD. The addition of 228°F to the current result should bound the thermal/hydraulic uncertainty associated with this scenario. This is true because the most important component in the DCD analysis PCT uncertainty adder is the reflood phase PCT increase associated with the uncertainty of the code itself, as established during the licensing of the large-break LOCA best-estimate methodology (Reference A-22). Because the code uncertainty term dominates, the PCT adder is not a strong function of the AP1000 DECLG calculated transient behavior. Moreover, the PCT sensitivity to variabilities in thermal/hydraulics at the low calculated PCT of the spurious ADS-4 actuation case is judged to be lower in magnitude than the sensitivity that applies at the much higher cladding temperature level of the DCD large-break LOCA analysis. Therefore, for the spurious ADS-4 actuation case, the PCT with uncertainties considered can be conservatively equated to the PCT as calculated by WCOBRA/TRAC plus 228°F, or 833 + 228 = 1061°F.

A5.6 T/H Uncertainty Analysis for Long-Term Cooling

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

Two cases of LTCC following a DEDVI line break are analyzed. These cases were determined by T/H uncertainty evaluations performed for AP1000 (in Section A5). One of these cases considers that the containment is isolated (Case F), and the other case considers that the containment isolation has failed (Case G). It is conservatively assumed that the DEDVI line break occurs in the PXS-B room. Since the size of this room is bigger than PXS-A, it reduces the containment water level during recirculation. It also takes more time for the water to fill it to the DVI nozzle elevation, where water can start flowing into the downcomer through the broken DVI line. In both cases, the general assumptions and methodology of the calculations are essentially the same. Conservative boundary and initial conditions are applied consistent with these multiple failure PRA-based scenarios to ensure that the T/H uncertainties contained within the success criteria are bounded.

A short summary follows of the two T/H uncertainty cases described herein.

- Case F:
  - DEDVI LOCA in line B
  - Available equipment – 1/1 CMT (A), both IRWST injection lines open with 1/2 valves open in each, only 1 recirculation line available with both valves open and this is the line attached to DVI-B, 3/4 ADS-4, PCS water drain with 1/3 valves open
  - Unavailable equipment – no ADS 1/2/3, PRHR, RNS injection/spill, IRWST gutter
  - Containment isolation is assumed to have worked.
- Case G:
  - DEDVI LOCA in line B
  - Available equipment – 1/1 CMT (A), both IRWST injection lines open with 1/2 valves open in each, 1/2 recirculation lines open with both valves open (line B), 4/4 ADS-4, PCS water drain with 1/3 valves open
  - Unavailable equipment – no ADS 1/2/3, PRHR, RNS injection/spill, IRWST gutter
  - Containment isolation is assumed to have failed (18-inch HVAC line remains open).

A5.6.1 WCOBRA/TRAC LTCC Modeling Methodology

The simulation methodology used in the current analyses is essentially the same as the one used for the AP600 design certification process (Reference A-4).

- The T/H uncertainty analyses are performed using the WCOBRA/TRAC thermal hydraulic computer code (Reference A-27).
- The WCOBRA/TRAC AP1000 model is the same as the one used in the AP1000 Post-LOCA Long-Term Cooling analysis (Reference A-26)
- The AP1000 LTCC simulations are performed using WCOBRA/TRAC in a transient mode. The transient mode approach has been validated by the Oregon State University Tests and was used in the AP600 Design Certification (Reference A-4).
- For each case, the AP1000 initial and boundary conditions are provided by a MAAP4 calculation. MAAP4 is capable of simulating the behavior and the interaction between the AP1000 primary system, the passive safety systems, the containment, and the containment systems – a feature that is not present in WCOBRA/TRAC.

- Like the MAAP4, the WCOBRA/TRAC simulation is performed with the following conservative general assumptions:
  - 102-percent core power
  - Appendix K decay heat
  - Maximum hydraulic resistance of the passive safety systems

A5.6.2 Methodology Implementation

The transient mode calculation using WCOBRA/TRAC allows simulation of long transients with reasonable computer resources. As was shown in the validation of methods used in the DCD analysis (Reference A-26), the calculation may be initiated from an arbitrary set of initial conditions. After an initial period of 500 to 1000 seconds, the plant reaches a quasi-steady-state that depends mostly on the system boundary conditions. During this "steady-state" period, the boundary conditions are kept constant. After that, they are set as a function of time depending on the time window being simulated.

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

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

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

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

A5.6.2.1 Case F – DEDVI Line Break in the PXS-B Room with Three of Four ADS Stage 4, Containment Isolated

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

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.

A7 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

A-10 AP600 Test and Analysis Plan for Design Certification, WCAP-14141

A-11 AP600 Emergency Response Guidelines

A-12 AP600 Emergency Response Guidelines Background Information

A-13 AP600 Implementation of the Regulatory Treatment of Nonsafety-Related Systems Process, WCAP-13856

A-14 AP600 Passive System Reliability Roadmap, NSD-NRC-96-4996, 8/9/96

A-15 NUREG-1512, "Final Safety Evaluation Report Related to Certification of the AP600 Standard Design," September 1998

A-16 LOCTA-IV Program, Loss of Coolant Transient Analysis, WCAP-8301, June 1974 (Westinghouse Proprietary)

A-17 Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code, WCAP-10054-P-A, August 1985 (Westinghouse Proprietary)

A-18 WCOBRA-TRAC OSU Long Term Cooling Final Validation Report, WCAP-14776, 11/96 (Westinghouse Proprietary)

A-19 MAAP4 Modular Accident Analysis Program, User's Manual, Rev. 0, May 1994

A-20 AP600 ATWS Analysis, SAE-APS-98-11, 1/22/98

A-21 "AP1000 PIRT and Scaling Assessment," WCAP-15613 (Proprietary) and WCAP-15706 (Non-Proprietary), March 2001

A-22 "Code Qualification Document for Best Estimate LOCA Analysis," WCAP-12945 (Westinghouse Proprietary), March 1998

A-23 "WCOBRA/TRAC Applicability to AP600 Large-Break Loss-Of-Coolant Accident," WCAP-14171 (Westinghouse Proprietary), March 1998

A-24 "AP1000 Code Applicability Report," WCAP-15644 (Westinghouse Proprietary) and WCAP-15707 (Non-Proprietary), May 2001

A-25 "Final Safety Evaluation Report Related to Certification of the AP600 Standard Design," NUREG-1512, September 1998

A-26 AP1000 DCD, APP-GW-GL-700

A-27 WCAP-12945, "Code Qualification Document for Best Estimate Analysis," Volumes 1 through 5, Revision 1 (Westinghouse Proprietary)

Table A2.3-1

FULL ADS SUCCESS CRITERIA[1]

Event	PRHR HX - on		PRHR HX - off	
	CMT - on	CMT - off	CMT - on	CMT - off
	Accum - off	Accum - on	Accum - off	Accum - on
RCS Transients, Loss of Power, Station Blackout	None (2)	None (2)	Auto 1/2 ADS stage 2/3 and auto 3/4 ADS stage 4	Man 3/4 ADS stage 4
RCS Leak	Auto 3/4 ADS stage 4	Man 3/4 ADS stage 4	Auto 1/4 ADS stage 2,3 and auto 3/4 ADS stage 4	Man 3/4 ADS stage 4
SGTR	None (3)	None (3)	Auto 1/4 ADS stage 2,3 and auto 3/4 ADS stage 4	Man 3/4 ADS stage 4
Small LOCA	Auto 3/4 ADS stage 4	Man 3/4 ADS stage 4	Auto 1/4 ADS stage 2,3 and auto 3/4 ADS stage 4	Man 3/4 ADS stage 4
Medium LOCA	Auto 3/4 ADS stage 4	Man 3/4 ADS stage 4	Auto 3/4 ADS stage 4	(7)
Spurious ADS	(4)		Auto 3/4 ADS stage 4 (5)	
Large LOCA	(4)		Auto 3/4 ADS stage 4 (6)	

Notes:

1. Automatic ADS actuation is via the protection and safety monitoring system (PMS). Any automatic ADS actuation can also be performed manually via PMS or DAS.
2. Successful PRHR HX operation obviates need for ADS.
3. SGTR does not require ADS operation if PRHR HX operates and SGs are isolated.
4. Operation of PRHR HX has no effect on ADS success criteria, use "PRHR HX - off" success criteria.
5. Spurious ADS requires 1/2 accumulators and 1/2 CMT to work.
6. Large LOCA requires 2/2 accumulators and 1/2 CMT to work.
7. No credit is given for success for this case; the time available for operator action is short.

Table A2.3-2

PARTIAL ADS SUCCESS CRITERIA[1]

Event	PRHR HX - on		PRHR HX - off	
	CMT - on	CMT - off	CMT - on	CMT - off
	Acc - off	Acc - on	Acc - off	Acc - on
RCS Transients, Loss of Power, Station Blackout	None (2)	None (2)	Auto 2/4 ADS stage 2/3 or 1/4 ADS stage 4	Man 2/4 ADS stage 2,3 or 1/4 ADS stage 4
RCS Leak	Auto 2/4 ADS stage 2/3 or 1/4 ADS stage 4	Man 2/4 ADS stage 2,3 or 1/4 ADS stage 4	Auto 2/4 ADS stage 2/3 or 1/4 ADS stage 4	Man 2/4 ADS stage 2,3 or 1/4 ADS stage 4
SGTR	None (3)	None (3)	Auto 2/4 ADS stage 2/3 or 1/4 ADS stage 4	Man 2/4 ADS stage 2,3 or 1/4 ADS stage 4
Small LOCA	(4)	(4)	Auto 2/4 ADS stage 2/3 or 1/4 ADS stage 4	Man 2/4 ADS stage 2,3 or 1/4 ADS stage 4
Medium LOCA	(4)	(4)	Auto 2/4 ADS stage 2/3 or 1/4 ADS stage 4	Man 2/4 ADS stage 2,3 or 1/4 ADS stage 4
Spurious ADS	None (5)	None (5)	None (5)	None (5)
Large LOCA	None (5)	None (5)	None (5)	None (5)

Notes:
1. Automatic ADS actuation is via PMS. Any automatic ADS actuation can also be performed manually via PMS or DAS.
2. Successful PRHR HX operation obviates need for ADS.
3. SGTR does not require ADS operation if PRHR HX operates and SGs are isolated.
4. Operation of PRHR HX has no effect on ADS success criteria.
5. These LOCAs are large enough to depressurize the RCS to allow RNS pumped injection.

Table A3.1-1 (Sheet 1 of 2)

PRA LOCA SIZE DEFINITIONS

Category	Basis for Minimum Size	Required RCS Pressure	AP1000 Min
RCS Leak	Leak that causes plant to shutdown	na	1 gpm
Small LOCA	Capacity of 1 CVS makeup pump	na	100 gpm 3/8"
Medium LOCA	RCS depres. to ADS 4 pres. interlock without any ADS	< 1200 psia	2"
Large LOCA	Break that requires 2 accumulators	na	9"

Note:
1. LOCA sizes are shown in inches ID.

Table A3.1-1 (Sheet 2 of 2)

INITIATING EVENTS WITH ADS ACTUATION

Event Tree	Alpha Designator for Success Paths	Grouping of Initiating Events Based on Plant Response
Large LOCA (LLOCA)	LLO	>9" LOCAs
Spurious ADS (SPADS)		
Medium LOCA (MLOCA)	MLO	2" to 9" LOCAs
CMT Line Break	CMT	
DVI Line Break	SIL	
Small LOCA (SLOCA)	SLO	< 2" LOCAs and High Pressure Events
RCS Leak	Transfer to SLOCA	
PRHR Tube Rupture	Transfer to SLOCA	
SG Tube Rupture	SGR	
Transients with Main FW	TRA	
Loss of Reactor Coolant Flow	Transfer to MLOCA (Stuck open pressurizer safety valve) Transfer to SLB (Stuck open SG safety valve)	
Loss of Main FW to 1 SG		
Power Excursion		
Loss of CCW/SWS		
Loss of Main FW to Both SGs		
Loss of Condenser		
Loss of Compressed Air		
Loss of Offsite Power	LSP SBO	
ATWS	ATW	
Steamline Break Downstream of MSIVs	SLB	
Steamline Break Upstream of MSIVs		
Stuck-open Secondary Side Safety Valve		

Table A3.1-2

SUCCESS PATHS WITH ADS ACTUATION
(Except Large LOCAs, LLOCA & SPAD)

	IRWST Injection	RNS Injection	
	Full ADS	**Full ADS**	**Partial ADS**
Automatic ADS	Section A3.2.1	MLO-OK1	Section A3.2.2
CMT injection	MLO-OK2		MLO-OK4
No accumulator injection	MLO-OK3	CMT-OK1	
	CMT-OK2		CMT-OK4
	CMT-OK3		
	SIL-OK1	SLO-OK1	
	SLO-OK2	SLO-OK5	SLO-OK4
	SLO-OK3		SLO-OK8
	SLO-OK6		
	SLO-OK7	SGR-OK3	
	SGR-OK4	TRA-OK4	SGR-OK3
	TRA-OK5	LSP-OK4	TRA-OK4
	LSP-OK5		LSP-OK4
	SBO-OK2	SLB-OK5	
	SLB-OK6		SLB-OK5
Manual ADS	Section A3.3.1		Section A3.3.2
No CMT injection	MLO-OK6	MLO-OK5	
Accumulator injection	MLO-OK7	CMT-OK5	MLO-OK8
	CMT-OK6		CMT-OK8
	CMT-OK7		
	SIL-OK2	SLO-OK9	
	SLO-OK10	SLO-OK13	SLO-OK12
	SLO-OK11		SLO-OK16
	SLO-OK14		
	SLO-OK15	SGR-OK5	
	SGR-OK6	TRA-OK6	SGR-OK5
	TRA-OK7	LSP-OK6	TRA-OK6
	LSP-OK7		LSP-OK6
	SBO-OK3	SLB-OK3	
	SLB-OK4	SLB-OK7	SLB-OK3
	SLB-OK8		SLB-OK7

Key to success path designators:
MLO = Medium LOCA
CMT = CMT line break
SIL = SI (DVI) line break
SLO = Small LOCA
SGR = SGTR
TRA = Transient
LSP = Loss of Offsite Power
SBO = Station Blackout (Loss of offsite power with loss of diesel generators)
SLB = Steam line break