19.59 PRA Results and Insights

19.59.1 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.

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 sealless reactor coolant pumps. 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.

19.59.2 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.

19.59.3 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 19.59-1. Figure 19.59-1 illustrates the relative contributions to core damage frequency from the various at-power initiating events. Table 19.59-2 shows the conditional core damage probability of the initiating events. The conditional core damage probability listed in Table 19.59-2 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 contributes 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.

19.59.3.1 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 19.59-3.

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 19.59-3. The top 25 cutsets for these sequences are given in Tables 19.59-4 through 19.59-13.

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.

19.59.3.2 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.

19.59.3.3 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.

19.59.3.4 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 19.59-14. 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 19.59-14, 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.

19.59.3.5 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.

19.59.3.6 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, or 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.

19.59.3.7 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.

19.59.3.8 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.

19.59.3.9 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.

19.59.4 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.

19.59.4.1 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

19.59.4.2 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.

19.59.5 Core Damage and Severe Release Frequency from Events at Shutdown

19.59.5.1 Summary of Shutdown Level 1 Results

As shown by the dominant cutsets of the AP600 and AP1000 shutdown models, 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 RNS during drained condition, and loss of offsite power during drained condition. The AP1000 and AP600 initiating event core damage contributions are similar for the two plants.

The dominant sequences are described in the subsections that follow. The dominant accident sequences comprise 95.3 percent of the level 1 shutdown PRA 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 76.7 percent of the CDF
- Loss of RNS initiating event during drained condition with a contribution of 10.4 percent of the CDF
- Loss of offsite power initiating event during drained condition with a contribution of 8.2 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 component cooling water system (CCS) or service water system (SWS) during mid-loop/vessel flange operation, which has an estimated duration of 120 hours per 18 months refueling cycle.

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

- Hardware failures of both service water pumps or common cause failure of digital input/output modules from the protection and monitoring system (PMS)
- Common cause failure of the ADS 4th stage squib valves
- Common cause failure of the recirculation line 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 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 cycle.

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

- Common cause failure of the RNS pumps to run
- Common cause failure of the recirculation line squib valves
- 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 cycle. 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:

- 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
- Failure of Ovation digital output modules for RNS-V055
- Common cause failure of the ADS 4th stage squib valves
- Common cause failure of batteries IDSA-DB-1A/1B
- Common cause failure to start engine-driven fuel pumps
- 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 cycle. 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:

- Failure of the RNS pump to run or restart
- Common cause failure of the ADS 4th stage squib valves
- Failure of Ovation digital output modules for RNS-V055
- Common cause failure of the recirculation line 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

Conclusions

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

- The overall shutdown core damage frequency is very small (1.03E-07/year).

- Initiating events during reactor coolant system drained conditions contribute approximately 95 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 is the initiating event with the greatest contribution (approximately 77 percent of the shutdown core damage frequency).

- Common cause failures of in-containment refueling water storage tank components contribute approximately 56 percent of the total shutdown core damage frequency. Common cause failure of the in-containment refueling water storage tank valves contributes approximately 45 percent of the total shutdown core damage frequency.

- Common cause failures of the automatic depressurization system stage 4 squib valves contribute approximately 26 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 22 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.

19.59.5.2 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 1.72E-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.

19.59.5.3 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 frequency and the low large release frequency.

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

19.59.6.1 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 are 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.

Internal flooding events during shutdown operations are also evaluated. A quantitative internal flooding PRA of AP1000 design performed to estimate plant CDF and LRF for at-power and during low-power and shutdown events provided the following results:

	Plant CDF	Plant LRF
Internal Flooding During At-Power Events	8.82E-10/yr	7.14E-11/yr
Internal Flooding During Low-Power and Shutdown Events	3.22E-09/yr	5.37E-10/yr

The minimization of potential flooding sources in the safety-related areas, in addition to the physical separation of redundant safety-related components and systems from each other and from nonsafety-related components, reduces the consequences of internal flooding. The core damage and large release frequencies arising from flooding events during shutdown operations are not appreciable contributors to overall AP1000 risk.

19.59.6.2 Results of Internal Fire Assessment

The total at-power, fire-induced core damage frequency is 5.61E-08 per reactor year. The estimated LRF is 4.54E-09/yr. 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.

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, using the focused PRA models, 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.

19.59.6.3 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.

19.59.7 Plant Dose Risk from Release of Fission-Products

The design certification of the AP1000 included consideration by the NRC of the topic referred to in this section.

19.59.8 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 19.59-17.

The contribution of various events to the at-power core damage frequency is shown in Figure 19.59-1.

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 19.59-17. 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.

19.59.9 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

19.59.9.1 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 sealless reactor coolant pumps, thus avoiding seal loss-of-coolant accident issues related to shaft seals 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.

19.59.9.2 Systems Design

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

19.59.9.2.1 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, and 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. This increases 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 it 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, the 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.

19.59.9.2.2 Nonsafety-Related Systems

The AP1000 has nonsafety-related systems capable of mitigating accidents. These systems use redundant components, which are powered by offsite and onsite power supplies. The 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 limited role in the plant risk profile because the passive safety-related systems do not require cooling, and the 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 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.

19.59.9.3 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. This eliminates 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.

19.59.9.4 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.

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

19.59.9.5 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 a severe accident.

19.59.9.5.1 Containment Isolation and Leakage

Failure of the containment isolation system before a severe accident will lead to a direct release pathway from the containment volume to the environment. The 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 been used only 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, the AP1000 provides significant protection against the failure to isolate the containment and against the failure of isolation valves to fully close.

19.59.9.5.2 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, and thereby provide 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.

19.59.9.5.3 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.

19.59.9.5.4 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.

19.59.9.5.5 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. This eliminates 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 prevent 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.

19.59.9.5.6 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, which 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 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.

19.59.9.5.7 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 low. The frequency of containment failures due to hydrogen combustion events is low given the high reliability of the hydrogen igniters.

19.59.9.5.8 Fission-Product Removal

The 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, which 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.

19.59.10 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).

19.59.10.1 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 that are 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 Section 17.4.

19.59.10.2 PRA Input to Tier 1 Information

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

19.59.10.3 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 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.

19.59.10.4 Summary of PRA Based Insights

The use of the PRA in the design process is discussed in subsection 19.59.2. A summary of the overall PRA results is provided in subsections 19.59.3 through 19.59.8. A discussion of the

AP1000 plant features important to reducing risk is provided in subsection 19.59.9. PRA-based insights are developed from this information and are summarized in Table 19.59-18.

19.59.10.5 Combined License Information

The Combined License applicant referencing the AP1000 certified design will confirm that the Seismic Margin Assessment analysis documented in Section 19.55 is applicable to the COL site. This will include a confirmation that the COL site seismic demand based on the site GMRS is enveloped by the Certified Seismic Design Response Spectra (CSDRS) seismic demand as defined by Tier 1 criteria for SSE as well as an assessment that no site specific effects such as seismically induced liquefaction settlements, slope stability, foundation failure, and relative displacements have the potential to lower the HCLPF values calculated for the certified design. Further evaluation will be required if the COL site is shown to be outside of the bounds of the SMA analysis documented in Section 19.55.

The Combined License holder 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 prior to fuel load. A verification walkdown will be performed with the purpose of identifying differences between the as-built plant and the design. Any differences will be evaluated to determine if there is a 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 holder.

The Combined License holder referencing the AP1000 certified design should compare the as-built SSC HCLPFs to those assumed in the AP1000 seismic margin evaluation prior to fuel load. Deviations from the HCLPF values or assumptions in the seismic margin evaluation due to the as-built configuration and final analysis should be evaluated to determine if vulnerabilities have been introduced. The requirements to which the equipment is to be purchased are included in the equipment specifications. Specifically, the equipment specifications include:

1. Specific minimum seismic requirements consistent with those used to define the Table 19.55-1 HCLPF values.

   This includes the known frequency range used to define the HCLPF by comparing the required response spectrum (RRS) and test response spectrum (TRS). The test response spectra must be chosen so as to demonstrate that no more than 1 percent rate of failure would be expected when the equipment is subjected to the applicable seismic margin ground motion for the equipment identified to be applicable in the Seismic Margin Insights of the Site-Specific PRA. The range of frequency response that is required for the equipment with its structural support is defined.

2. Hardware enhancements that were determined in previous test programs and/or analysis programs will be implemented.

The Combined License holder 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 19.59-18 prior to fuel load. 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.

Based on site-specific information, the COL should also reevaluate the qualitative screening of external events (PRA Section 58.1). If any site-specific susceptibilities are found, the PRA should be updated to include the applicable external event. The Combined License information requested in this subsection has been partially addressed in APP-GW-GLR-101 (Reference 19.59-4), and the applicable changes are incorporated into the DCD. Additional work is required by the Combined License applicant to address the aspects of the Combined License information requested in this subsection as delineated in the following paragraph:

> The Combined License applicant will confirm that the High Winds, Floods, and Other External Events analysis documented in Section 19.58 is applicable to the COL site. Further evaluation will be required if the COL site is shown to be outside of the bounds of the High Winds, Floods, and Other External Events analysis documented in Section 19.58.

The Combined License holder 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 prior to fuel load. 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 APP-GW-GL-027, "Framework for AP1000 Severe Accident Management Guidance," (Reference 19.59-2). The Combined License information requested in this subsection has been partially addressed in APP-GW-GLR-070 (Reference 19.59-1), and the applicable changes are incorporated into the DCD. APP-GW-GLR-070 closes the development portion of this COL item. Additional work is required by the Combined License applicant to address the aspects of the Combined License information requested in this subsection as delineated in the following paragraph:

> The Combined License applicant will implement the AP1000 Severe Accident Management Guidance from APP-GW-GLR-070 on a site-specific basis.

The Combined License holder referencing the AP1000 certified design will perform a thermal lag assessment of the as-built equipment listed in Tables 6b and 6c in Attachment A of APP-GW-GLR-069 (Reference 19.59-5) 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 performed prior to fuel load and is required only for equipment used for severe accident mitigation that has not been tested at severe accident conditions. The Combined License holder 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 19.59-3).

19.59.11 References

19.59-1 APP-GW-GLR-070, "Development of Severe Accident Management Guidance," Westinghouse Electric Company LLC.

19.59-2 APP-GW-GL-027, "Framework for AP1000 Severe Accident Management Guidance," Westinghouse Electric Company LLC.

19.59-3 "Large Scale Hydrogen Burn Equipment Experiments," EPRI-NP-4354, December 1985.

19.59-4 APP-GW-GLR-101, "AP1000 Probabilistic Risk Assessment Site Specific Considerations," Westinghouse Electric Company LLC.

19.59-5 APP-GW-GLR-069, "Equipment Survivability Assessment," Westinghouse Electric Company LLC.

Table 19.59-1

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 19.59-2

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 19.59-3 (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 19.59-3 (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 19.59-3 (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 19.59-3 (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	2emlo-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 19.59-4 (Sheet 1 of 3)

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

NUMBER	CUTSET PROB.	PERCENTAGE	BASIC EVENT NAME		
1	5.09E-08	74.04	SAFETY INJECTION LINE BREAK INITIATING EVENT OCCURS	2.12E-04	IEV-SI-LB
			IRWST 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 19.59-4 (Sheet 2 of 3)

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

NUMBER	CUTSET PROB.	PERCENTAGE	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 19.59-4 (Sheet 3 of 3)

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

NUMBER	CUTSET PROB.	PERCENTAGE	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 19.59-5

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

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

Table 19.59-6 (Sheet 1 of 3)

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

NUMBER	CUTSET PROB.	PERCENTAGE	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 19.59-6 (Sheet 2 of 3)

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

NUMBER	CUTSET PROB.	PERCENTAGE	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 19.59-6 (Sheet 3 of 3)

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

NUMBER	CUTSET PROB.	PERCENTAGE	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 19.59-7 (Sheet 1 of 3)

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

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

Table 19.59-7 (Sheet 2 of 3)

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

NUMBER	CUTSET PROB.	PERCENTAGE	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 19.59-7 (Sheet 3 of 3)

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

NUMBER	CUTSET PROB.	PERCENTAGE	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 19.59-8

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

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

Table 19.59-9 (Sheet 1 of 3)

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

NUMBER	CUTSET PROB.	PERCENTAGE	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 19.59-9 (Sheet 2 of 3)

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

NUMBER	CUTSET PROB.	PERCENTAGE	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 19.59-9 (Sheet 3 of 3)

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

NUMBER	CUTSET PROB.	PERCENTAGE	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 19.59-10 (Sheet 1 of 3)

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

NUMBER	CUTSET PROB.	PERCENTAGE	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 19.59-10 (Sheet 2 of 3)

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

NUMBER	CUTSET PROB.	PERCENTAGE	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 19.59-10 (Sheet 3 of 3)

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

NUMBER	CUTSET PROB.	PERCENTAGE	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 19.59-11 (Sheet 1 of 3)

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

NUMBER	CUTSET PROB.	PERCENTAGE	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 19.59-11 (Sheet 2 of 3)

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

NUMBER	CUTSET PROB.	PERCENTAGE	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 19.59-11 (Sheet 3 of 3)

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

NUMBER	CUTSET PROB.	PERCENTAGE	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 19.59-12 (Sheet 1 of 3)

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

NUMBER	CUTSET PROB.	PERCENTAGE	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 19.59-12 (Sheet 2 of 3)

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

NUMBER	CUTSET PROB.	PERCENTAGE	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

Table 19.59-12 (Sheet 3 of 3)

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

NUMBER	CUTSET PROB.	PERCENTAGE	BASIC EVENT NAME		
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
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 19.59-13 (Sheet 1 of 3)

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

NUMBER	CUTSET PROB.	PERCENTAGE	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 19.59-13 (Sheet 2 of 3)

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

NUMBER	CUTSET PROB.	PERCENTAGE	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 19.59-13 (Sheet 3 of 3)

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

NUMBER	CUTSET PROB.	PERCENTAGE	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 19.59-14

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	
250 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 19.59-15

SUMMARY OF AP1000 PRA RESULTS

Events	Core Damage Frequency (per year)		Large Release Frequency (per year)	
	At-Power	**Shutdown**	**At-Power**	**Shutdown**
Internal Events	2.41E-07	1.03E-07	1.95E-08	1.72E-08
Internal Flood	8.82E-10	3.22E-09	7.14E-11	5.37E-10
Internal Fire	5.61E-08	8.5E-08[(1)]	4.54E-09	1.43E-08
Sum =	2.97E-07	1.91E-07	2.41E-08	3.20E-08

Note:

1. Internal fire during shutdown is evaluated quantitatively as a response to an NRC question and is not reported elsewhere in this document.

Table 19.59-16 not used.

Table 19.59-17

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%

Note:

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 19.59-18 (Sheet 1 of 25)

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.	Tier 1 Information
	- 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.	
	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.	6.3.2 & 7.3
	- 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.	

Table 19.59-18 (Sheet 2 of 25)

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:	19.36
- High pressure melt ejection	
- Direct containment heating	
- Induced steam generator tube rupture	
- Induced RCS piping rupture and rapid hydrogen release to containment	

Table 19.59-18 (Sheet 3 of 25)

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 19.59-18 (Sheet 4 of 25)

AP1000 PRA-BASED INSIGHTS

Insight	Disposition
1d. IRWST subsystem provides a safety-related means of performing the following functions:	6.3
- 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:	
IRWST subsystem has the following flowpaths:	Tier 1 Information
- 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.	
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 19.59-18 (Sheet 5 of 25)

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 19.59-18 (Sheet 6 of 25)

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 19.59-18 (Sheet 7 of 25)

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 initiation 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 19.59-18 (Sheet 8 of 25)

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: - 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.	Tier 1 Information
- The DAS automatic actuation signals are generated in a functionally diverse manner from the PMS signals. Diversity between DAS and PMS is achieved by the use of different architectures, different hardware implementations, and different software, if any. Software diversity between the DAS and PMS will be achieved through the use of different algorithms, logic, program architecture, executable operating system, and executable software/logic.	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

Table 19.59-18 (Sheet 9 of 25)

AP1000 PRA-BASED INSIGHTS

Insight	Disposition
3. (cont.)	
The DAS reactor trip function is to trip the control rods by deenergizing the motor-generator set.	7.7.1.11
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
The DAS manual actuation cables are located within the nuclear island and, therefore, are protected from external hazards, such as high winds.	
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 algorithms 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

Table 19.59-18 (Sheet 10 of 25)

AP1000 PRA-BASED INSIGHTS

Insight	Disposition
5a. (cont.)	
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
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 250 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

Table 19.59-18 (Sheet 11 of 25)

AP1000 PRA-BASED INSIGHTS

Insight	Disposition
5b. (cont.) The Class 1E batteries are included in the D-RAP.	17.4
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, 3, and 4 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

Table 19.59-18 (Sheet 12 of 25)

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
Recognizing the increased vulnerability to risk with the plant in a “drained” condition, when the refueling cavity is not full and PRHR HXs are not available, entry into this condition and time spent in this condition during anticipation of a potentially severe high wind event will be minimized.	13.5
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

Table 19.59-18 (Sheet 13 of 25)

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 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

Table 19.59-18 (Sheet 14 of 25)

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
- Administrative controls are established to maintain the performance of the fire protection system.	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
Administrative controls are established to maintain the performance of the fire protection system.	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 room 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. Differences between the as-built plant and the basis for the AP1000 seismic margin analysis are reviewed.	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 address 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 50.44).	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. Severe accident management guidance is developed and implemented using the suggested framework provided in APP-GW-GL-027.	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. Severe accident management guidance is developed and implemented using the suggested framework provided in APP-GW-GL-027.	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. Containment spray is not credited in the PRA. Failure of the nonsafety-related containment spray does not prevent the plant achieving the safety goals. Severe accident management guidance for operation of the nonsafety-related containment spray system is developed and implemented using the suggested framework provided in APP-GW-GL-027.	6.5.2 19.59 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. These valves are identified as being risk-significant SSCs and are included in the D-RAP. Operability of DAS for selected containment isolation actuations is addressed by short-term availability controls.	17.4 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. 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 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. Severe accident management guidance for venting containment is developed and implemented using the suggested framework provided in APP-GW-GL-027.	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.	Differences between the as-built plant and the design used as the basis for the AP1000 PRA and Table 59-18 are reviewed. 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. Based on site-specific information, the qualitative screening of external events (PRA Section 58.1) is evaluated. If any site-specific susceptibilities are found, the PRA should be updated to include the applicable external event.	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.	Procedures to control transient combustibles are established.	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. Procedures and training must be developed to 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. Procedures and training must be developed to 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. Administrative controls to ensure that inadvertent opening of this valve is unlikely must be developed.	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
On a source range flux doubling signal, the PMS automatically closes two safety-related CVS makeup line isolation valves, closes two safety-related CVS demineralized water suction valves to the makeup pumps, and trips the makeup pumps. On a reactor trip or low input voltage to the Class 1E dc power system battery chargers, the PMS closes the two safety-related CVS demineralized water suction valves to the makeup pumps and aligns the makeup pump suction to the boric acid tank.	7.3.1.2.14
71. Procedures will be prepared 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% 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

Table 19.59-18 (Sheet 24 of 25)

AP1000 PRA-BASED INSIGHTS

Insight	Disposition
73. A 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
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
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 are established.	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
79. Generic open items and plant-specific action items resulting from NRC review of the I&C platform are resolved.	7.1.6
80. An analysis is provided that demonstrates that operator actions, which minimize the probability of the potential for spurious ADS actuation as a result of a fire, can be accomplished within 30 minutes following detection of the fire and the procedure for the manual actuation of the valve to allow fire water to reach the automatic fire system in the containment maintenance floor.	9.5.1.8
81. Procedures to minimize risk when fire areas are breached during maintenance are established. These procedures will address a fire watch for fire areas breached during maintenance.	9.5.1.8
82. It is important to maintain the low-temperature overpressure protection provided by the RNS relief valve to ensure that the reactor vessel pressure and temperature limits are not exceeded during shutdown conditions. Isolation of the RNS and its relief valve is permitted during shutdown conditions in case the hot legs empty due to a loss of RCS inventory; if the RNS is isolated, an alternate vent path would be opened, such as the ADS Stage 1, 2, and 3 valves.	16.1 (LCO Basis 3.4.14)

Table 19.59-18 (Sheet 25 of 25)

AP1000 PRA-BASED INSIGHTS

Insight	Disposition
83. The passive cooling system louvers and screens cover openings located all around the containment, into an enclosed volume where the air inlet ducts are located. The screens are designed to help prevent foreign objects or debris from entering the air flow path. In the event of a snow or ice storm, some fraction of these air inlets can become blocked with snow or ice. The results of analysis, made available to the staff during the design certification of the AP1000, show that a considerable fraction of the inlet area can be blocked without a significant effect on the peak containment pressure for design basis events.	
Louvers are arranged within the air inlets to minimize the entrance of debris into the inlets. These louvers are fixed and, therefore, will not block the air flow path.	
The chimney outlet is designed to produce the necessary air flow in the event of an accident. The outlet contains two heavy grates to guard against missiles, and it is fully screened to prevent foreign objects from entering the containment annulus. The presence of a positive air flow during normal operation helps prevents ice and snow from entering the chimney.	6.2.2.2.4
Air-only cooling of the containment provides cooling necessary to maintain containment integrity with a high level of confidence for the first 24 hours following an accident in the event there is no water cooling from PCS.	
There is a surveillance requirement (SR 3.6.6.5) to verify every 24 months that the air flow path is unobstructed.	3.6.6
84. The AP1000 is protected against external floods up to the 100-foot level, which corresponds to the ground level at each plant. From this point, the ground is graded so that water naturally flows away from the plant structures.	
85. The plant is designed such that the 100-foot level is slightly above grade and the level of anticipated external flooding. Below grade is protected against flooding by a water barrier consisting of waterstops and a waterproofing system. Seismic Category I SSCs below grade are designed to withstand hydrostatic pressures.	
The seismic Category I SSCs below grade are protected against external flooding by a water barrier consisting of waterstops and a waterproofing system.	3.4.1.1.1
86. The vacuum relief system is important for the integrity of the containment during an event where a vacuum is developed inside containment. The vacuum relief system consists of redundant relief devices, and its function is to prevent differential pressure between the inside and outside of the containment from exceeding the design value.	

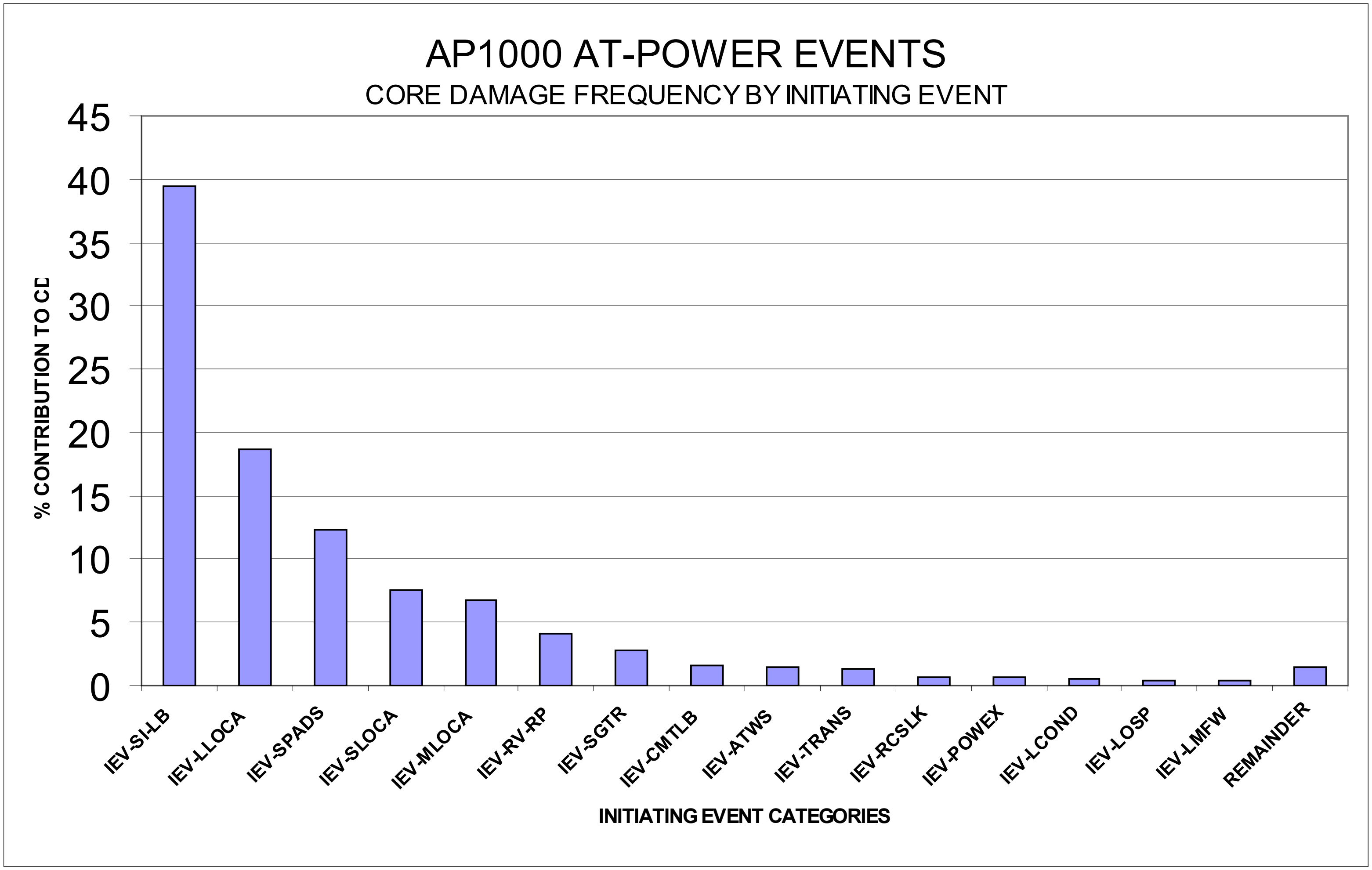

Figure 19.59-1

Contribution of Initiating Events to Core Damage

Figure 19.59-2 not used.