17 RESULTS SUMMARY

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17 RESULTS SUMMARY

17.1 OVERVIEW OF RESULTS AND INSIGHTS

The objectives of Sections 17, 18 and 19 are to provide insights about the design of the ESBWR, and to ensure that PRA modeling assumptions are preserved such that they remain valid for plants referencing the certified design. Sections 17, 18 and 19 follow an integrated approach to analyzing the PRA results. The PRA results encompass the entire quantified risk profile, which includes internal and external events (seismic events are treated separately in a margins analysis), severe accident progression analysis, and offsite consequence analysis for at-power and shutdown operating conditions. Risk-significant insights and assumptions are identified from the systematic evaluations of Sections 17 and 18. The significant results and assumptions are assigned to one of three categories: Operational Program, Design Requirement, or PRA Insight.

Operational programs include operator actions that are considered to be significant. These are summarized in Table 17.1-3, and they are incorporated into the Human Factors Engineering process.

Design requirements capture key PRA system modeling information. In most cases, the system models are based upon actual design features, as described in the DCD. However, some design information has not yet been developed. This information is treated as design assumptions, and must be preserved to ensure that the PRA model reflects the to-be-built plant. Design Requirements are listed in Table 18-1.

PRA insights are retained within the design engineering organization. These are key insights that describe significant aspects of the PRA model or results. PRA insights can be consulted in PRA model updates, risk-informed applications, and expert panel applications. PRA insights are listed in Table 18-1.

The objective of Section 19 is to develop preventive maintenance recommendations for risk-significant components. The recommendations are based on generic information and are used to demonstrate the ability to ensure that the assumptions in the PRA model relative to the reliability of components can be preserved throughout the operating life of the plant.

17.1.1 Objectives

Section 17 compiles the quantification results of the PRA models to identify the dominating contributions to the overall risk profile and includes insights from the seismic margins analysis. These results are evaluated to identify features that are responsible for maintaining low risk, and those that perform significant mitigation functions. Such analyses provide important information about:

(1) Areas where certain design features are the most effective in reducing risk with respect to the design of operating reactors;

(2) Major contributors to risk, such as hardware failures and human errors;

(3) Major contributors to maintaining the “built-in” plant safety and ensuring that the risk does not increase unacceptably;

(4) Major contributors to the uncertainty associated with the risk estimates; and

(5) Sensitivity of risk estimates to uncertainties associated with failure data, assumptions made in the PRA models, lack of modeling details in certain areas, and previously raised issues. The evaluations account for uncertainties in parameters, along with sensitivity studies to ensure that the PRA parameters are a reasonable representation of the design and operation of the ESBWR.

The following subsections contain the significant results and insights of the comprehensive ESBWR PRA. This information is used in an expert panel process to identify risk-significant SSCs that are part of the Reliability Assurance Program described in DCD Tier 2 Section 17.4. Table 17.1-1 is a summary of the CDF and LRF results for each quantification. NEDO-33411 contains the basic events from the ESBWR PRA models that are considered risk-significant based on the criteria discussed above. Table 17.1-3 is a list of risk-significant operator actions.

17.1.2 Defining Significance

17.1.2.1 Background

PRA results can be classified as dominant or significant relative to a particular point of reference. While dominant risk contributors are identified relative to a specific measure, such as an accident sequence, risk-significance is defined relative to the NRC Safety Goals. In order to determine the thresholds for risk significance, the ASME Standard for Probabilistic Risk Assessment for Nuclear Power Plant Applications (Reference 17-.1) provides the following definitions for significance:

- Significant Basic Event - A basic event that has an F-V importance greater than 0.005 or a RAW importance greater than 2.
- Significant Cutset - One of the set of cutsets that, when rank-ordered by decreasing frequency, aggregate to a specified percentage of the CDF (or LRF), or that individually contribute more than a specified percentage of CDF (or LRF.)

For the significant cutset threshold, the ASME standard recommends an aggregate percentage of 95% and an individual percentage of 1%. Cutset significance may be measured relative to overall CDF (or LRF) or relative to an individual accident sequence CDF (or LRF.)

Additionally, the ASME standard defines significance on a broader scale:

- Significant Accident Sequence - One of the set of accident sequences, defined at the functional or systemic level, that, when rank-ordered by decreasing frequency, aggregate to a specified percentage of the CDF, or that individually contributes more than a specified percentage of CDF.
- Significant Accident Progression Sequence - One of the set of accident sequences, contributing to large release frequency that, when rank-ordered by decreasing frequency, aggregate to a specified percentage of the LRF, or that individually contribute more than a specified percentage of LRF.
- Significant Containment Challenge - A containment challenge that results in a containment failure mode that is represented in a significant accident progression sequence.

For the accident sequences, the specified percentage that the ASME standard recommends is an aggregate percentage of 95% and an individual percentage of 1%.

The risk thresholds that are recommended by the ASME standard provide a reasonable point of reference for existing plants, whose internal events CDF values range from roughly 1 E-4/yr to 1 E-6/yr. However, some of the advanced LWR CDF values are more than two orders of magnitude lower. Applying the current recommended risk thresholds is not reasonable because it represents risk that is substantially lower than the safety goals. For example, for an advanced BWR with an at-power internal events CDF of 2 E-8/yr using a RAW importance of 2as a threshold for risk significance would lead one to conclude that changes in CDF of 2 E-8/yr are risk significant. This is clearly an unreasonable threshold for determining risk significance.

If the ASME standard for a significant cutset, (aggregate 95 percent of the CDF and individual 1%) is used for the ESBWR internal events PRA, then over 30,000 cutsets would be significant. The contribution of cutset number 30,153, which aggregates to 95%, is 1.9 E-14/year, which is an individual contribution of 0.0002%. In other words, virtually every event would be significant, relative to a safety goal of 1 E-4/yr including cutsets whose individual contributions are statistically insignificant.

17.1.2.2 ESBWR Risk Significance

The criteria for categorization into low and high significance are related to the RG 1.174 (Reference 17-2) acceptance criteria for changes in CDF and LERF (which is equivalent to LRF for the purpose of this analysis.) Thus, the risk significance criteria are a function of the base case CDF and LERF rather than being fixed for all plants. The RG 1.174 acceptance criteria for CDF and LERF relate changes in risk to baseline risk. For example, a change in CDF of < 1E-6/yr is considered to be not risk significant for a plant with a baseline CDF of < 1 E-5/yr. The LERF criteria are one order of magnitude lower.

Risk significance for the ESBWR is thus defined in terms of risk increase (RAW) and risk contribution (F-V). An increase in CDF risk of greater than or equal to 1 E-7/year is considered risk significant for the design certification ESBWR PRA. This is clearly below the Region III limit in Figures 3 and 4 of RG 1.174, that is, the lowest level of regulatory concern. The upper limit of Region III is 1 E-6/year for CDF changes. This significant margin is considered sufficient to address potential cumulative effects. Thus, a CDF increase of 1E-7/year relative to the internal events CDF of 1.22E-8/year would yield a RAW value of (1.12 E-7 / 1.22E-8) = 9.2. An individual contribution of 1E-9/year is considered to conservatively represent the threshold for risk contribution. This translates to an F-V value of (1E-9 / 1.22E-8) = 0.08.

For the purpose of this section, the following conservative thresholds are used to identify potentially risk-significant information:

- F-V greater than or equal to 0.01
- RAW greater than or equal to 5.0 for individual events
- RAW greater than or equal to 50 for common cause failures

These thresholds are applied to the PRA models with CDF values of approximately 1 E-8/year. For the PRA models with CDF values of approximately 1 E-9/year or lower, the thresholds are increased by a factor of 10. For example, the High Winds CDF is 1.3 E-9/year and the thresholds for risk significance are 0.1 for F-V and 50 for RAW, including common cause failures.

17.2 AT-POWER INTERNAL EVENTS

This subsection contains the majority of the results and discussions. The at-power internal events PRA is the most highly developed analysis and involves SSCs and operator actions that are most prevalent for operating reactors. This model tends to provide the most comprehensive lists of CDF and LRF insights. As such, a majority most of the results, dominant contributors, and insights are derived from this analysis. The other analyses are compared against the at-power internal events analysis to identify additional significant results.

17.2.1 Significant Core Damage Sequences – At Power Internal Events

The CDF for at-power internal events is 1.22 E-8/year. The top ten at-power internal event sequences are described below, on a functional level. This distillation of the PRA accident sequences is the important insights that represent the behavior of the ESBWR design in response to postulated accidents.

(1) T-IORV063

Inadvertent Opening of a Relief Valve

Scram is successful

High Pressure Injection fails

Depressurization is successful

Low Pressure Injection fails

(2) AT-T-GEN023

General Transient with ATWS

Scram fails

SLCS fails

(3) T-FDW050

Loss of Feedwater

Scram is successful

Isolation Condensers fail

Depressurization is successful

Low Pressure Injection fails

(4) T-IORV018

Inadvertent Opening of a Relief Valve

Scram is successful

High Pressure Injection fails

Active Low Pressure Injection fails

Depressurization fails

(5) AT-T-GEN021

General Transient with ATWS

Scram fails

One or more SRVs sticks open

Failure to maintain RPV water level

(6) T-IORV065

Inadvertent Opening of a Relief Valve

Scram is successful

High Pressure Injection fails

Depressurization fails

(7) AT-T-LOPP013

Loss of Preferred Power with ATWS

Scram fails

One or more SRVs stick open

Failure to maintain level

(8) T-IORV017

Inadvertent Opening of a Relief Valve

Scram is successful

High Pressure Injection fails

Depressurization is successful

Low Pressure Injection fails

(9) LL-S-FWB045

Line Break in Feedwater Line B

Scram is successful

LOCA depressurizes RPV – fails Feedwater

Low Pressure Injection fails

(10) T-LOPP050

Loss of Preferred Power

Scram is successful

Isolation Condensers fail

Depressurization is successful

Low Pressure Injection fails

The dominant sequences typically do not contain independent component failures. Instead, they consist of common cause failures that disable entire mitigating functions. And, it is important to note that multiple mitigating functions must fail in the dominant sequences, so a single common cause event is insufficient to directly result in core damage. The ATWS sequences are dominated by an assumed failure of the control rods to insert into the core due to mechanical binding. Core damage in ATWS accident sequences results from the inability to maintain a lowered RPV water level prior to achieving subcriticality.

Table 17.2-1 identifies the top initiating events as: %T-IORV, Inadvertent Opening of a Safety Relief Valve; T-GEN, General Transient; %T-FDW, Loss of Feedwater; %T-LOPP, Grid-Related Loss of Preferred Power; and %LL-S-FDWB, Large Steam LOCA in Feedwater Line B. The initiating event frequency for %T-IORV considers the likelihood of a stuck open Safety/Relief Valve (SRV), an inadvertent opening or closing of an SRV, and the spurious opening of two vent valves in series from the Isolation Condensers to the suppression pool. ESBWR relief valve lifting after transient is less likely due to the ability of ICS to maintain the reactor at high pressure conditions without reaching the SRV relief setpoints. This frequency is conservative because transient-induced stuck open SRVs are handled in the event trees following a transient initiating event. A lower frequency could be justified; however, there are no risk-significant insights identified. As discussed in Section 2, the initiating event frequencies for %T-GEN, %T-FDW, and %T-LOPP are applied using industry operating experience data. ESBWR values are expected to be lower. LOCA frequencies (%LL-S-FDWB) are also applied using operating experience data. Overall, none of the dominant initiating events are considered to have unique risk insights.

17.2.2 Significant Functions, SSCs, and Operator Actions – At-Power Internal Events

Tables 17.2-2 through 17.2-4 list the significant components for at-power internal events. The F-V importance values in Table 17.2-2 are low, which indicates that the risk profile is balanced and does not contain dominating contributors to risk. The most important systems, based on RAW importance values, are the 6.9 kV AC PIP buses. Loss of a PIP bus during at-power operation would result in a plant trip and loss of one division of active mitigation systems. Other important components include SLCS valves and accumulators, and feedwater runback logic load drivers that support ATWS response. These findings are consistent with the top accident sequences discussed in section 17.2.1. The top common cause failures are the mechanical binding of control rods, wetwell to drywell vacuum breakers and their isolation valves, RPS scram valves, digital I&C software, depressurization squib valves, and GDCS check valves and squib valves. These common cause failures correspond to failures of RPS and passive functions. In each case, at least one diverse backup system is available.

A sensitivity study in Section 11 evaluates the risk importance values on a system level, with the following results:

Systems With F-V > 0.1

E50 Gravity Driven Cooling System

N21 Condensate & Feedwater System

B21 Automatic Depressurization System

C71 Reactor Protection System

C12 Control Rod Drive System

C63 Safety-Related Distributed Control and Instrumentation

G21 Fuel & Auxiliary Pool Cooling System

R21 Standby On Site AC Power Supply

C41 Standby Liquid Control System

Systems With RAW > 50

T10 Containment System

R13 Uninterruptable AC Power Supply System

C71 Reactor Protection System

B21 Automatic Depressurization System

C63 Safety-Related Distributed Control and Instrumentation

B32 Isolation Condenser System

R16 Direct Current Power Supply

E50 Gravity Driven Cooling System

T15 Passive Containment Cooling System

R11 Medium Voltage Distribution System

C62 Nonsafety-Related Distributed Control and Instrumentation

P41 Plant Service Water System

C72 Diverse Protection System

R10 Electrical Power Distribution System

Table 17.2-5 shows that the important operator actions involve recognizing the need for depressurization or providing low pressure injection in particular scenarios; failure to restart feedwater pumps during certain ATWS scenarios; and pre-initiator valve mispositioning events in the FAPCS, CRD, and RCCW systems. Information on important operator actions is incorporated into the human factors engineering program.

17.2.3 Large Release Sequences –At Power Internal Events

The at-power internal events large release frequency (LRF) is 9.62 E-10/year.

17.2.3.1 Significant Large Release Sequences –At Power Internal Events

The most important release sequences are core damage sequences that challenge the containment but only result in controlled releases (i.e., not "large releases".) They are transient-initiated events, followed by either a loss of low pressure injection, or a loss of high pressure injection with a failure to depressurize. The most important "large release" sequences are not risk significant because their frequencies are significantly lower than the NRC Safety Goals. They are initiated by transients or LOCAs and they result in conservatively postulated ex-vessel explosion scenarios that lead to containment failure.

17.2.3.2 Significant Large Release - At-Power Internal Initiating Events

The most important Level 2 initiating events are %T-IORV, %T-GEN, and %T-FDW; however, they result in controlled releases. The most important large release initiating event is %LL-S-FDWB, which represents a Large LOCA in Feedwater Line B. This is due to the impact the line has on mitigating functions that become disabled.

17.2.3.3 Significant Large Release Functions, SSCs, and Operator Actions –At-Power Internal Events

The most important systems and components that are unique to large release sequences are the RWCU/SDC isolation valves and the BiMAC device. The most important operator actions that are unique to large releases involve failure to actuate shutdown cooling during an ATWS, failure to vent the containment, and valve mispositioning events on FAPCS and CRD manual valves.

The following large release basic events are significant based on the RAW criterion:

- Common Cause Failures of SLU, OLU, and DTM components in C7,
- Operator Actions.

17.2.3.4 Containment Performance – At-Power Internal Events

The potential for containment failure due to combustible gas generation, containment bypass and overpressurization was evaluated. In addition, the frequency of containment failure events due to the phenomenological events discussed in Section 21 (CCIW, CCID, DCH, EVE) was determined. Because of the ESBWR design and reliability of containment systems, the most likely containment response to a severe accident is associated with successful containment isolation, vapor suppression and containment heat removal. As a result, the containment provides a highly reliable barrier to the release of fission products after a severe accident, with the dominant release category being that defined by Technical Specification leakage (TSL).

A containment penetration screening evaluation indicated that there are only a few penetrations that required isolation to prevent significant offsite consequences. All potential leakage paths feature multiple containment isolation valves. Thus, the probability of the bypass failure mode is dominated by common cause hardware failures, resulting in a calculated frequency of containment bypass about three orders of magnitude lower than the TSL release category.

The conditional containment failure probability is 7.9%, which meets the recommended goal of less than 10%.

17.2.4 Significant Offsite Consequences – At-Power Internal Events

The Level 3 results indicate that the offsite consequences due to internal at-power events are negligible. The results, including sensitivity studies, demonstrate that the estimated offsite consequences are less than the defined individual, societal, and radiation dose limits by several orders of magnitude.

This conclusion applies to the at-power external events also. However, the shutdown events are assumed to result in a direct containment bypass.

17.2.5 Summary of Important Results and Insights – At-Power Internal Events

The dominant sequences typically do not contain multiple independent component failures. Instead, they consist of common cause failures that disable entire mitigating functions. And, it is important to note that multiple mitigating functions must fail in the dominant sequences, so a single common cause event is insufficient to directly result in core damage. The ATWS sequences are dominated by an assumed failure of the control rods to insert into the core due to mechanical binding. Core damage in ATWS accident sequences results from the inability to maintain a lowered RPV water level prior to achieving subcriticality.

Important operator actions in the dominant accident sequences are manual depressurization and the ability to provide low pressure injection.

17.3 AT-POWER EXTERNAL EVENTS

Risks due to external events initiated by fire, flood, and high winds are quantified and described below. Seismic risk is assessed qualitatively using a seismic margins analysis. Other external events are treated in a bounding manner such that potential ESBWR sites are able to reach the same conclusions for external events.

17.3.1 Significant Core Damage Sequences – Fire

The CDF for fire initiated events is 8.1 E-9/year. This is a screening value that is appropriately used in the design phase to evaluate spatial interactions. It does not credit automatic or manual suppression, and it conservatively assumes that each fire creates maximum damage. The most important fire sequences involve loss of one electrical division due to a fire in the Reactor Building, with common cause software failures on the digital control systems, and failure of GDCS injection due to common cause failure of check valves, and failure of the Operators to recognize the need to provide low pressure injection.

17.3.2 Significant Initiating Events – Fire

The most important fire-initiated events involve fires in the Reactor Building that disable Division I or II electrical equipment. This disables one train of mitigating systems; however, the other train is typically not affected by the fire. A fire in the Switchyard is important due to its effect of causing a loss of preferred power.

17.3.3 Significant Functions, SSCs, and Operator Actions – Fire

The ESBWR, due to its basic layout and safety design features, is inherently capable of mitigating potential internal fires. Safety system redundancy and physical separation by fire barriers ensure that in all cases a single fire limits damage to a single safety system division or system redundancy. Fire propagation to neighboring areas presents a relatively minor risk contribution.

Fires in the control room are assumed to affect the execution of human actions from there. One feature relevant to the design is that a fire in the control room does not affect the automatic actuations of the safety systems. Additionally, the existence of remote shutdown panels allows the opportunity to perform manual actuations for failed automatic actuations that may occur.

Similar to the internal events analysis, the F-V importance values for fires are low, which indicates a balanced risk profile. The most important component failures are the ICS vent valves failing to open.

17.3.4 Large Release Sequences – Fire

The LRF for fire-initiated events is 4.61 E-10/year. One additional significant function is the ability of the BiMAC device to cool the molten core; however, the value of the failure probability is a conservative point estimate. There are no additional insights from SSCs, or Operator Actions in large release fire sequences.

17.3.5 External Events - Flood

17.3.5.1 Significant Core Damage Sequences - Flood

The CDF for flooding initiated events is 1.62 E-9/year. A conservative approach was used because plant layout details are not yet fully developed.

17.3.5.2 Significant Initiating Events – Flood

The flood initiating events with the highest importance values are leaks initiated in the Service Water Pumphouse that disable Service Water Train A or B. The next most important event is a Circulating Water pipe break in the Turbine Building. The most important flood in the Reactor Building is in the Control Rod Drive Room due to a CRD pipe break.

17.3.5.3 Significant Functions, SSCs, and Operator Actions - Flood

There are no additional functions, SSCs or operator actions that are significant solely due to flooding events.

17.3.6 Large Release Sequences – Flood

17.3.6.1 Significant Large Releases - Flood

The LRF due to flooding is 2 E-10/year. There are no additional functions, SSCs or operator actions that are significant solely due to flooding events.

17.3.7 External Events – Wind

The CDF due to high winds from tornadoes or hurricanes is 1.34 E-9/year.

17.3.7.1 Significant Core Damage Sequences - Wind

The most important wind sequences are similar to internal event %T-LOPP sequences, with the additional loss of the condensate storage tank in some sequences.

17.3.7.2 Significant Initiating Events - Wind

The internal events initiating event %T-LOPP-WR is used to reflect the assumption that a damaging high wind event will resemble a loss of preferred power.

17.3.7.3 Significant Functions, SSCs, and Operator Actions – Wind

There are no additional functions, SSCs or operator actions that are significant solely due to high wind events.

17.3.8 Large Release Sequences – Wind

Technical specifications leakage (TSL) is the Level 2 success state and indicates an intact, controlled containment boundary. The high winds analysis produces a total non-TSL release frequency of 1.22E-09. The at-power non-TSL release frequency is 3.00E-11/yr. The shutdown non-TSL release frequency is 1.19E-09, the same as the shutdown CDF. This is because the containment is assumed to be open during all Mode 5 and Mode 6 operations. This leads to containment bypass for all shutdown sequences.

17.3.8.1 Containment Performance – Wind

The plant should not be in a Mode 6 Unflooded condition when a hurricane strike occurs. This is because in Mode 6 unflooded the containment is open, the reactor vessel is open and the water above the core will not keep the core cool for an extended period of time.

17.3.9 Summary of Important Results and Insights – At-Power External Events

The ESBWR high wind analysis explicitly quantifies accident sequences initiated by tornado winds. Straight winds are lesser velocity winds that pose minimal challenges to the plant design. Hurricane winds are quantified using a bounding analysis. Due to the strength of construction of the ESBWR Category I buildings, the effects of a tornado strike are limited to Loss of Preferred Power events with a potential loss of the Condensate Storage Tank. Overall risk from tornados and high winds is further minimized by design features such as the diesel driven fire protection pump for alternate RPV injection, and the DC batteries with a 72-hour operational life.

17.4 SHUTDOWN INTERNAL EVENTS

17.4.1 Significant Core Damage Sequences – Shutdown Internal Events

The CDF for shutdown internal events is 9.37 E-9/year.

17.4.1.1 Initiating Events – Shutdown Internal Events

The most important initiating events are LOCAs in instrument lines below the top of the fuel, during Mode 5 and Mode 6.

17.4.1.2 Component Reliability and Availability – Shutdown Internal Events

Additional significant basic events based on the RAW criterion are the GDCS squib valves that supply the deluge function to the BiMAC device. Additional Operator Actions include failure to close the drywell hatches after a pipe break in an RWCU drain line or in an instrumentation line below the top of active fuel.

17.4.1.3 Results – Shutdown Internal Events

The greatest contribution to shutdown risk comes from breaks in lines connected to the vessel below TAF. In these cases, the lower drywell equipment hatch or personnel hatch is likely to be open to facilitate work in the lower drywell. Although the frequency of these events is very low, there is only one method for mitigation – manual closure of the hatches.

The next largest contributions to shutdown risk are due to losses of preferred power (LOPP) or loss of all service water (PSWS) during Mode 6 Unflooded. These scenarios are higher than other shutdown scenarios due to ICS being unavailable in Mode 6.

LOCA events above the TAF do not contribute much at all to the entire overall shutdown CDF. The highest sequence for any LOCA above the fuel has a value of 3E-13, which is four orders of magnitude below the highest cutset, and two orders of magnitude below the highest non-LOCA cutsets.

17.4.2 Significant Large Release Sequences – Shutdown Internal Events

During shutdown conditions all core damage sequences are assumed to be bypass sequences, therefore, large release sequences are not evaluated further

.

17.5 SHUTDOWN EXTERNAL EVENTS

17.5.1 Significant Core Damage Sequences

The CDF for fire-initiated events during shutdown is 2.71 E-8/year. The CDF for flood-initiated events during shutdown is 5.24 E-9/year. The CDF for high wind-initiated events during shutdown is 1.19 E-9/year.

17.5.1.1 Initiating Events

The most important fire-initiated Shutdown Events are Loss of RWCU/SDC due to fire in Turbine Building – Modes 5 and 6-Unflooded; and, Loss of service water due to fire in Service Water Pumphouse - Mode 6-Unflooded.

The most important flood-initiated Shutdown Events are a Break in GDCS Pool A or D during Mode 6-Unflooded; Reactor Building CRD Room Line Break; and, Service Water Building SW Line Break.

17.5.1.2 Component Reliability and Availability

For fire-initiated events, the most important components are the CRD pumps and valves in the high pressure injection path. Also, the diesel fire pump injection flow path is marginally important for these sequences. For flood-initiated events, there are no additional components that are significant in these scenarios.

17.5.1.3 Operator Actions

There are two additional operator actions that are important for Shutdown external events, and the both involve fire-initiated events. They involve failure to initiate CRD injection, and failure to use the diesel makeup pump in the LPCI mode of FAPCS.

17.5.2 Summary of Important Shutdown Results and Insights

Administrative controls should be used to ensure that the Drywell Hatches can be closed during shutdown conditions if a loss of coolant event is initiated in the drywell.

The ESBWR plant has excellent capability to mitigate the consequences due to postulated internal fires. The separation criteria incorporated in the ESBWR design, especially for the safety-related systems and RTNSS systems, greatly enhance the redundancy and ensure that a single fire cannot defeat a whole system.

The dominant risk contributors with respect to fire scenarios are the postulated fires in turbine building general area and the plant service water area. As stated in Section 12.8.4, conservatism is embedded in the shutdown modeling by assuming a fire in these two fire areas can induce a shutdown initiating event and fails all the components in the subject fire area. With the exception of large turbine building fires, both fire areas are well accessible for fire suppression. Therefore, the shutdown fire risk analysis proves that the robustness of ESBWR plant against the postulated fires.

The dominant risk contributor with respect to shutdown modes is “Mode 6 Unflooded.” This is consistent with the baseline shutdown CDF results since the isolation condenser system is not credited in the Mode 6 Unflooded event trees. Therefore, it is necessary to ensure the operability of the systems critical to decay heat removal function during this mode.

Several GDCS system (E50) CCF basic events, pre-initiator and post-initiator operation actions contribute significantly to the fire shutdown CDF.

17.6 INSIGHTS FROM SEISMIC MARGINS ANALYSIS

A PRA-based seismic margins analysis is performed for the ESBWR to calculate high confidence low probability of failure (HCLPF) accelerations for important accident sequences and accident classes. The ESBWR seismic margins HCLPF accident sequence analysis concludes that the ESBWR is inherently capable of safe shutdown in response to beyond design basis earthquakes and has a plant level HCLPF of 1.67 times the safe shutdown earthquake (SSE). Chapter 15 of NEDO-33201 provides details of the PRA-based seismic margin assessment.

DCD Tier 2 Table 19.2-4 contains the systems evaluated in the ESBWR and contains minimum HCLPF ratio for these systems.

17.6.1 Significant Core Damage Sequences

A PRA-based Seismic Margins Analysis is used to derive seismic vulnerability insights. Therefore, there are no CDF calculations performed. The Seismic Margins Analysis concludes that the most significant HCLPF sequences are seismic-induced loss of DC power and seismic-induced ATWS due to seismic-induced failure of the fuel channels and seismic-induced failure of the SLC tank.

Based on previous industry seismic analyses, seismic risk is dominated by seismic-induced SSC failures, and not by random SSC failures or human actions. Human actions are typically not necessary until the long-term.

There are no additional SSCs that are risk significant due to the Seismic Margins Analysis.

17.6.2 Significant Large Release Sequences

A PRA-based Seismic Margins Analysis is used to derive seismic vulnerability insights. Therefore, there are no LRF calculations performed.

17.6.3 Significant Offsite Consequences

A PRA-based Seismic Margins Analysis is used to derive seismic vulnerability insights. Therefore, there are no off-site consequences calculations performed. Due to the bounding method that is used to calculate the seismic margin, it is considered to be unnecessary to extrapolate offsite consequences.

17.7 SUMMARY OF IMPORTANT RESULTS AND INSIGHTS

Table 17.1-1 lists the CDF and LRF values for each quantified risk condition in the ESBWR PRA. It is important to restate that because the individual CDF values are developed with differing levels of conservatism, it is not meaningful to add CDF or LRF values to create total values.

Table 17.1-2 lists risk-significant SSCs based on the results of this section. This table includes a cross-reference for each basic event and the conditions in which it is significant.

Table 17.1-3 lists risk-significant operator actions based on the results of this section. Tables 17.2-1 through 17.2-5 contain the results of the at-power internal events importance analyses.

17.8 REFERENCES

17-1 ASME RA-Sb-2005, "Standard for Probabilistic Risk Assessment for Nuclear Power Plant Applications," Addendum B to ASME RA-S-2002, ASME, New York, New York, December 30, 2005.

17-2 US NRC, Regulatory Guide 1.174, "An Approach For Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," Revision 1, November 2002

17-3 NEDO 33411, Risk Significance of Structures, Systems and Components For the Design Phase of the ESBWR, Revision 0, March 2008

Table 17.1-1

Results

	Internal Events	Fire	Flood	High Winds
At-Power CDF	1.22E-08	8.06E-09	1.62E-09	1.34E-09
Shutdown CDF	9.37E-09	2.71E-08	5.24E-09	1.19E-09
At-Power LRF	9.62E-10	5 E-10	2 E-10	3 E-11
Shutdown LRF	9.37E-09	2.71E-08	5.24E-09	1.19E-09

Table 17.1-2

ESBWR Structures, Systems and Components That Are Risk Significant DELETED

Table 17.1-3

ESBWR Risk-Significant Operator Actions

Basic Event	Description
C12-BV_-RE-F013A	MISPOSITION OF VALVE F013A
C12-BV_-RE-F013B	MISPOSITION OF VALVE F013B
C12-BV_-RE-F015A	MISPOSITION OF VALVE F015A
C12-BV_-RE-F015B	MISPOSITION OF VALVE F015B
C12-BV_-RE-F021A	MISPOSITION OF VALVE F021A
C12-BV_-RE-F021B	MISPOSITION OF VALVE F021B
C12-BV_-RE-F065	MISPOSITION OF LOCKED OPEN VALVE F065
C12-XHE-FO-LEVEL2	OPERATOR FAILS TO BACK-UP CRD ACTUATION
DWH-1	CLOSE LOWER DRYWELL HATCH
DWH-2	FAILURE TO CLOSE DRYWELL HATCH
G21-BV_-RE-F308	MISPOSITION OF VALVE F308
G21-BV_-RE-F334	MISPOSITION OF VALVE F334
G31-XHE-FO-SDC	OPERATOR FAILS TO ACTUATE SDC MODE NO MSL LOCA OUSIDE CONTAINMENT NO SLCS
N21-XHE-FO-FWRERUN	OPERATOR FAILS TO RESTART FDW AFTER RUNBACK - ATWS
P21-BV_-RE-F049A	MISPOSITION OF RCCW INLET TO CRD HEAT EXCHANGER
P21-BV_-RE-F049B	MISPOSITION OF RCCW INLET TO CRD HEAT EXCHANGER
P21-BV_-RE-F050A	MISPOSITION OF RCCW OUTLET FROM CRD HEAT EXCHANGER
P21-BV_-RE-F050B	MISPOSITION OF RCCW OUTLET FROM CRD HEAT EXCHANGER
P54-XHE-FO-REOPEN	OPERATOR FAILS TO RECOGNIZE OPENING OF F009 OR F026
REC_MANSCRAM	CREDIT FOR MANUAL SCRAM
R-M6-G31	FAILURE TO RECOVER RWCU/SDC
R-PSWS-6	SERVICE WATER RECOVERY
U43-XHE-FO-LPCI	OPERATOR FAILS TO ACTUATE U43 IN LPCI MODE
XXX-XHE-FO-DEPRESS	OPERATOR FAILS TO RECOGNIZE NEED OF DEPRESSURIZATION
XXX-XHE-FO-LPMAKEUP	OP. FAILS TO RECOG. NEED FOR LOW PRESS. MAKEUP AFTER DEPRESSURIZATION

Table 17.2-1
At-Power Internal Initiating Events

	Sequence Designator	Description	% CDF	Cumulative
1	%T-IORV	Inadvertent Open SRV	36.50%	36.50%
2	%T-GEN	General Transient	18.40%	54.9%
3	%T-FDW	Loss of Feedwater	16.70%	71.6%
4	%T-LOPP-GR	Loss of Preferred Power - Grid Faults	6.03%	77.6%
5	%LL-S-FDWB	Large LOCA in FW Line B	4.35%	82.0%
6	%T-LOPP-SC	Loss of Preferred Power – Switchyard Faults	3.34%	85.3%
7	%T-PCS	Loss of Power Conversion System	3.03%	88.4%
8	%T-IA	Loss of Instrument Air	1.99%	90.3%
9	%T-LOPP-WR	Loss of Preferred Power – Weather Related	1.53%	91.9%
10	%ML-L	Medium LOCA – Liquid Break	1.37%	93.2%
11	%SL-S	Small LOCA – Steam Break	1.20%	94.4%
12	%LL-S	Large LOCA – Steam Break	0.97%	95.4%

Table 17.2-2

At-Power Internal Events F-V Importance

Event Name	Fus Ves	Description
R21-DG_-FR-DGA	3.02E-02	DIESEL GENERATOR "A" FAILS TO RUN GIVEN START
R21-DG_-FR-DGB	2.97E-02	DIESEL GENERATOR "B" FAILS TO RUN GIVEN START
C41-UV_-CC-F004A	2.60E-02	CHECK VALVE F004A FAILS TO OPEN
C41-UV_-CC-F004B	2.60E-02	CHECK VALVE F004B FAILS TO OPEN
C41-UV_-CC-F005A	2.60E-02	CHECK VALVE F005A FAILS TO OPEN
C41-UV_-CC-F005B	2.60E-02	CHECK VALVE F005B FAILS TO OPEN
N21-ACV-CC-F0016	1.94E-02	AIR OPERATED VALVE F0016 FAILS TO OPEN

Table 17.2-3

At-Power Internal Events RAW Importance

Event Name	Ach W	Description
R11-BAC-LP-100B3	383.28	6.9 KV AC PIP-A BUS 1000B3 FAILS DURING OPERATION
R11-BAC-LP-100A3	61.49	6.9 KV AC PIP-A BUS 1000A3 FAILS DURING OPERATION
C41-ACV-OC-F002A	33.5	AIR OPERATED VALVE F002A FAILS TO REMAIN OPEN
C41-ACV-OC-F002B	33.5	AIR OPERATED VALVE F002B FAILS TO REMAIN OPEN
C41-ACV-OC-F002C	33.5	AIR OPERATED VALVE F002C FAILS TO REMAIN OPEN
C41-ACV-OC-F002D	33.5	AIR OPERATED VALVE FAILS TO REMAIN OPEN
C41-UV_-CC-F004A	33.48	CHECK VALVE F004A FAILS TO OPEN
C41-UV_-CC-F004B	33.48	CHECK VALVE F004B FAILS TO OPEN
C41-UV_-CC-F005A	33.48	CHECK VALVE F005A FAILS TO OPEN
C41-UV_-CC-F005B	33.48	CHECK VALVE F005B FAILS TO OPEN
C41-TNK-RP-A001A	33.47	ACCUMULATOR A001A FAILS CATASTROPHICALLY
C41-TNK-RP-A001B	33.47	ACCUMULATOR A001B FAILS CATASTROPHICALLY
C72-LDD-FC-FWRB1	30.3	LOAD DRIVER FAILS TO ENERGIZE FWRB CIRCUIT
C72-LDD-FC-FWRB2	30.3	LOAD DRIVER FAILS TO ENERGIZE FWRB CIRCUIT
R16-BT_-LP-R16BTA3	27.82	BATTERY R16-BTA3 FAILS TO PROVIDE OUTPUT
R16-LCB-CO-FROMR16BTA3	26.78	CIRCUIT BREAKER FROM R16-BTA3 OPENS SPURIOUSLY
R16-LCB-CO-R16A3SWGR1	26.78	CIRCUIT BREAKER 1 FROM R16-A3 OPENS SPURIOUSLY
R16-LCB-CO-R16A3SWGR2	26.78	CIRCUIT BREAKER 2 FROM R16-A3 OPENS SPURIOUSLY
N21-ACV-OC-F0018	26.61	AIR OPERATED VALVE N21-F018 FAILS TO REMAIN OPEN
P22-ACV-OC-F0006	26.61	TCCW HX FLOW CONTROL VALVE FAILS TO REMAIN OPEN
R16-BDC-LP-R16A3	25.57	DC BUS R16-A3 FAILS DURING OPERATION
P22-TNK-RP-A001	24.4	TCCW SURGE TANK LEAKS CATASTROPHICALLY
P30-TNK-RP-A001	24.14	CST LEAKS CATASTROPHICALLY
P22-ACV-CO-F0008	22.55	TCCW HX BYPASS VALVE FAILS TO REMAIN CLOSED
N21-ACV-CC-F0016	10.65	AIR OPERATED VALVE F0016 FAILS TO OPEN
N21-ACV-OC-F0016	9.44	AIR OPERATED VALVE N21-F0016 FAILS TO REMAIN OPEN
P22-HX_-PG-B001A	8.85	HEAT EXCHANGER 1A FAILS
P22-HX_-PG-B001B	8.85	HEAT EXCHANGER 1B FAILS

Table 17.2-3

At-Power Internal Events RAW Importance

Event Name	Ach W	Description
P41-ACV-OC-F001A	8.85	VALVE P22-F001A FAILS TO REMAIN OPEN
P41-ACV-OC-F001B	8.85	VALVE P41-F001B FAILS TO REMAIN OPEN
P41-SYS-FC-HVACPSW-A	8.44	PSW-A ROOM COOLING FAILURE
P41-SYS-FC-HVACPSW-B	8.44	PSW-B ROOM COOLING FAILURE
P52-UV_-OC-F006	8.38	CHECK VALVE FAILS TO REMAIN OPEN
N21-MOV-OC-F0057	8.13	MOV N21-F0057 FAILS TO REMAIN OPEN
P52-TNK-RP-RCV002	7.76	RECEIVER TANK FAILS CATASTROPHICALLY
P22-MOV-OC-F0005A	7.3	MOV FOR HX 1A FAILS TO REMAIN OPEN
P22-MOV-OC-F0005B	7.3	MOV FOR HX 1B FAILS TO REMAIN OPEN
P41-MOV-OC-F008A	7.3	VALVE P41-F008A FAILS TO REMAIN OPEN
P41-MOV-OC-F008B	7.3	VALVE P41-F008B FAILS TO REMAIN OPEN
P52-BV_-OC-F004	6.12	MANUAL VALVE TRANSFERS CLOSED
P52-BV_-OC-F005	6.12	MANUAL VALVE TRANSFERS CLOSED
R10-SYS-FF-500KV	5.81	500KV SWITCHYARD FAILS DURING OPERATION
B21-UV_-CC-F102B	5.34	CHECK VALVE #1 IN FEEDWATER LINE B FAILS TO REOPEN
B21-UV_-CC-F103B	5.34	CHECK VALVE #2 IN FEEDWATER LINE B FAILS TO REOPEN
C12-UV_-CC-F022	5.34	CHECK VALVE F022 FAILS TO OPEN

Table 17.2-4

At-Power Internal Events Common Cause Failures

Event Name	Ach W	Description
C12-ROD-CF-SCRAM	1.31E+06	CCF OF CONTROL RODS TO INSERT
T10-UV_-CC-VBISVS_1_2_3	8.78E+03	CCF of three components: T10-UV_-CC-ISV1 & T10-UV_-CC-ISV2 & T10-UV_-CC-ISV3
T10-VB_-CC_1_2_3	8.76E+03	CCF of three components: T10-VB_-CC-VB1 & T10-VB_-CC-VB2 & T10-VB_-CC-VB3
C12-AOV-CF-SCRV126	5.24E+03	CCF TO OPEN OF AIR OPERATED SCRAM VALVE AOV-126
C63-CCFSOFTWARE_S	2.03E+03	Common cause failure of software, for spurious
B21-SQV-CC_ALL	1.30E+03	CCF of all components in group 'B21-SQV-CC'
E50-UV_OC_ALL	991.56	CCF of all components in group 'E50-UV_OC'
E50-SQV-CC_ALL	984.98	CCF of all components in group 'E50-SQV-CC'
C62-CCFSOFTWARE	395.9	Common cause failure of software
T15-FLT-PP_ALL	384.7	CCF of all components in group 'T15-FLT-PP'
P41-FAN-FR_ALL	382.77	CCF of all components in group 'P41-FAN-FR'
P41-STR-PG_ALL	382.18	CCF of all components in group 'P41-STR-PG'
P41-FAN-FR_1_2	382.1	CCF of two components: P41-FAN-FR-0001A & P41-FAN-FR-0001B
P41-FAN-FR_1_4	382.1	CCF of two components: P41-FAN-FR-0001A & P41-FAN-FR-0002B
P41-FAN-FR_2_3	382.1	CCF of two components: P41-FAN-FR-0001B & P41-FAN-FR-0002A
P41-FAN-FR_3_4	382.1	CCF of two components: P41-FAN-FR-0002A & P41-FAN-FR-0002B
P41-MPW-FR_ALL	381.37	CCF of all components in group 'P41-MPW-FR'
P41-FAN-FR_1_2_3	373.94	CCF of three components: P41-FAN-FR-0001A & P41-FAN-FR-0001B & P41-FAN-FR-0002A
P41-FAN-FR_1_2_4	373.94	CCF of three components: P41-FAN-FR-0001A & P41-FAN-FR-0001B & P41-FAN-FR-0002B
P41-FAN-FR_1_3_4	373.94	CCF of three components: P41-FAN-FR-0001A & P41-FAN-FR-0002A & P41-FAN-FR-0002B

Table 17.2-4

At-Power Internal Events Common Cause Failures

Event Name	Ach W	Description
C12-ROD-CF-SCRAM	1.31E+06	CCF OF CONTROL RODS TO INSERT
P41-FAN-FR_2_3_4	373.94	CCF of three components: P41-FAN-FR-0001B & P41-FAN-FR-0002A & P41-FAN-FR-0002B
P41-STR-PG_1_2_3	368.2	CCF of three components: P41-STR-PG-D01A & P41-STR-PG-D01B & P41-STR-PG-D02A
P41-STR-PG_1_2_4	368.2	CCF of three components: P41-STR-PG-D01A & P41-STR-PG-D01B & P41-STR-PG-D02B
P41-STR-PG_1_3_4	368.2	CCF of three components: P41-STR-PG-D01A & P41-STR-PG-D02A & P41-STR-PG-D02B
P41-STR-PG_2_3_4	368.2	CCF of three components: P41-STR-PG-D01B & P41-STR-PG-D02A & P41-STR-PG-D02B
C63-CCFSOFTWARE	348.25	Common cause failure of software
T15-HX_-PP_ALL	225.22	CCF of all components in group 'T15-HX_-PP'
C72-CCFSOFTWARE	176.4	COMMON CAUSE FAILURE OF DPS PROCESSORS
R13-INV-FC-CCFSR_ALL	175.38	CCF of all components in group 'R13-INV-FC-CCFSR'
R13-INV-FC-CCFNSR_ALL	159.49	CCF of all components in group 'R13-INV-FC-CCFNSR'
C72-LOG-FC-D_1_2_3	153.95	CCF of three components: C72-LOG-FC-D1DPS & C72-LOG-FC-D2DPS & C72-LOG-FC-D3DPS
C72-LOG-FC-D_1_2	147.18	CCF of two components: C72-LOG-FC-D1DPS & C72-LOG-FC-D2DPS
C72-LOG-FC-D_1_3	147.18	CCF of two components: C72-LOG-FC-D1DPS & C72-LOG-FC-D3DPS
C72-LOG-FC-D_2_3	147.18	CCF of two components: C72-LOG-FC-D2DPS & C72-LOG-FC-D3DPS
R16-BT_-LP-CCFNSR_ALL	122.19	CCF of all components in group 'R16-BT_-LP-CCFNSR'
R13-INV-FC-CCFNSR_1_3_5	115.65	CCF of three components: R13-INV-FC-R13A1 & R13-INV-FC-R13B1 & R13-INV-FC-R13C
C72-LDD-CF-LOADS	106.74	COMMON CAUSE FAILURE OF DPS LOAD DRIVERS
R10-XFH-LP-CCF_ALL	101.97	CCF of all components in group 'R10-XFH-LP-CCF'
R16-BT_-LP-CCFSR_ALL	97.15	CCF of all components in group 'R16-BT_-LP-CCFSR'
T15-FLT-PP_1_2_3	75.89	CCF of three components: T15-FLT-PP-D001A & T15-FLT-PP-D001B & T15-FLT-PP-D001C

Table 17.2-4

At-Power Internal Events Common Cause Failures

Event Name	Ach W	Description
C12-ROD-CF-SCRAM	1.31E+06	CCF OF CONTROL RODS TO INSERT
T15-FLT-PP_1_2_4	75.89	CCF of three components: T15-FLT-PP-D001A & T15-FLT-PP-D001B & T15-FLT-PP-D001D
T15-FLT-PP_1_2_5	75.89	CCF of three components: T15-FLT-PP-D001A & T15-FLT-PP-D001B & T15-FLT-PP-D001E
T15-FLT-PP_1_2_6	75.89	CCF of three components: T15-FLT-PP-D001A & T15-FLT-PP-D001B & T15-FLT-PP-D001F
T15-FLT-PP_1_3_4	75.89	CCF of three components: T15-FLT-PP-D001A & T15-FLT-PP-D001C & T15-FLT-PP-D001D
T15-FLT-PP_1_3_5	75.89	CCF of three components: T15-FLT-PP-D001A & T15-FLT-PP-D001C & T15-FLT-PP-D001E
T15-FLT-PP_1_3_6	75.89	CCF of three components: T15-FLT-PP-D001A & T15-FLT-PP-D001C & T15-FLT-PP-D001F
T15-FLT-PP_1_4_5	75.89	CCF of three components: T15-FLT-PP-D001A & T15-FLT-PP-D001D & T15-FLT-PP-D001E
T15-FLT-PP_1_4_6	75.89	CCF of three components: T15-FLT-PP-D001A & T15-FLT-PP-D001D & T15-FLT-PP-D001F
T15-FLT-PP_1_5_6	75.89	CCF of three components: T15-FLT-PP-D001A & T15-FLT-PP-D001E & T15-FLT-PP-D001F
T15-FLT-PP_2_3_4	75.89	CCF of three components: T15-FLT-PP-D001B & T15-FLT-PP-D001C & T15-FLT-PP-D001D
T15-FLT-PP_2_3_5	75.89	CCF of three components: T15-FLT-PP-D001B & T15-FLT-PP-D001C & T15-FLT-PP-D001E
T15-FLT-PP_2_3_6	75.89	CCF of three components: T15-FLT-PP-D001B & T15-FLT-PP-D001C & T15-FLT-PP-D001F
T15-FLT-PP_2_4_5	75.89	CCF of three components: T15-FLT-PP-D001B & T15-FLT-PP-D001D & T15-FLT-PP-D001E
T15-FLT-PP_2_4_6	75.89	CCF of three components: T15-FLT-PP-D001B & T15-FLT-PP-D001D & T15-FLT-PP-D001F
T15-FLT-PP_2_5_6	75.89	CCF of three components: T15-FLT-PP-D001B & T15-FLT-PP-D001E & T15-FLT-PP-D001F
T15-FLT-PP_3_4_5	75.89	CCF of three components: T15-FLT-PP-D001C & T15-FLT-PP-D001D & T15-FLT-PP-D001E
T15-FLT-PP_3_4_6	75.89	CCF of three components: T15-FLT-PP-D001C & T15-FLT-PP-D001D & T15-FLT-PP-D001F

Table 17.2-4

At-Power Internal Events Common Cause Failures

Event Name	Ach W	Description
C12-ROD-CF-SCRAM	1.31E+06	CCF OF CONTROL RODS TO INSERT
T15-FLT-PP_3_5_6	75.89	CCF of three components: T15-FLT-PP-D001C & T15-FLT-PP-D001E & T15-FLT-PP-D001F
T15-FLT-PP_4_5_6	75.89	CCF of three components: T15-FLT-PP-D001D & T15-FLT-PP-D001E & T15-FLT-PP-D001F
R16-BT_-LP-CCFNSR_1_3_5	67.35	CCF of three components: R16-BT_-LP-R16BTA1 & R16-BT_-LP-R16BTB1 & R16-BT_-LP-R1
B32-HX_-PG_1_2	66.95	CCF of two components: B32-HX_-PG-HX001A & B32-HX_-PG-HX001B
B32-HX_-PG_1_3	66.95	CCF of two components: B32-HX_-PG-HX001A & B32-HX_-PG-HX001C
B32-HX_-PG_1_4	66.95	CCF of two components: B32-HX_-PG-HX001A & B32-HX_-PG-HX001D
B32-HX_-PG_1_6	66.95	CCF of two components: B32-HX_-PG-HX001A & B32-HX_-PG-HX002B
B32-HX_-PG_1_7	66.95	CCF of two components: B32-HX_-PG-HX001A & B32-HX_-PG-HX002C
B32-HX_-PG_1_8	66.95	CCF of two components: B32-HX_-PG-HX001A & B32-HX_-PG-HX002D
B32-HX_-PG_2_3	66.95	CCF of two components: B32-HX_-PG-HX001B & B32-HX_-PG-HX001C
B32-HX_-PG_2_4	66.95	CCF of two components: B32-HX_-PG-HX001B & B32-HX_-PG-HX001D
B32-HX_-PG_2_5	66.95	CCF of two components: B32-HX_-PG-HX001B & B32-HX_-PG-HX002A
B32-HX_-PG_2_7	66.95	CCF of two components: B32-HX_-PG-HX001B & B32-HX_-PG-HX002C
B32-HX_-PG_2_8	66.95	CCF of two components: B32-HX_-PG-HX001B & B32-HX_-PG-HX002D
B32-HX_-PG_3_4	66.95	CCF of two components: B32-HX_-PG-HX001C & B32-HX_-PG-HX001D
B32-HX_-PG_3_5	66.95	CCF of two components: B32-HX_-PG-HX001C & B32-HX_-PG-HX002A
B32-HX_-PG_3_6	66.95	CCF of two components: B32-HX_-PG-HX001C & B32-HX_-PG-HX002B

Table 17.2-4

At-Power Internal Events Common Cause Failures

Event Name	Ach W	Description
C12-ROD-CF-SCRAM	1.31E+06	CCF OF CONTROL RODS TO INSERT
B32-HX_-PG_3_8	66.95	CCF of two components: B32-HX_-PG-HX001C & B32-HX_-PG-HX002D
B32-HX_-PG_4_5	66.95	CCF of two components: B32-HX_-PG-HX001D & B32-HX_-PG-HX002A
B32-HX_-PG_4_6	66.95	CCF of two components: B32-HX_-PG-HX001D & B32-HX_-PG-HX002B
B32-HX_-PG_4_7	66.95	CCF of two components: B32-HX_-PG-HX001D & B32-HX_-PG-HX002C
B32-HX_-PG_5_6	66.95	CCF of two components: B32-HX_-PG-HX002A & B32-HX_-PG-HX002B
B32-HX_-PG_5_7	66.95	CCF of two components: B32-HX_-PG-HX002A & B32-HX_-PG-HX002C
B32-HX_-PG_5_8	66.95	CCF of two components: B32-HX_-PG-HX002A & B32-HX_-PG-HX002D
B32-HX_-PG_6_7	66.95	CCF of two components: B32-HX_-PG-HX002B & B32-HX_-PG-HX002C
B32-HX_-PG_6_8	66.95	CCF of two components: B32-HX_-PG-HX002B & B32-HX_-PG-HX002D
B32-HX_-PG_7_8	66.95	CCF of two components: B32-HX_-PG-HX002C & B32-HX_-PG-HX002D
R10-XFH-LP-CCF_2_4	63.27	CCF of two components: R10-XFH-LP-RATB & R10-XFH-LP-UATB
B32-NPO-CC_ALL	50.86	CCF of all components in group 'B32-NPO-CC'

Table 17.2-5

At-Power Internal Events Operator Actions

Event Name	Fus Ves	Description
XXX-XHE-FO-DEPRESS	2.89E-01	OPERATOR FAILS TO RECOGNIZE NEED OF DEPRESSURIZATION
XXX-XHE-FO-LPMAKEUP	2.08E-01	OP. FAILS TO RECOG. NEED FOR LOW PRESS. MAKEUP AFTER DEPRESSURIZATION
G21-BV_-RE-F334	1.27E-01	MISPOSITION OF VALVE F334
C12-BV_-RE-F065	1.14E-01	MISPOSITION OF LOCKED OPEN VALVE F065
N21-XHE-FO-FWRERUN	7.19E-02	OPERATOR FAILS TO RESTART FDW AFTER RUNBACK – ATWS
C12-BV_-RE-F013A	2.55E-02	MISPOSITION OF VALVE F013A
C12-BV_-RE-F013B	2.55E-02	MISPOSITION OF VALVE F013B
C12-BV_-RE-F015A	2.55E-02	MISPOSITION OF VALVE F015A
C12-BV_-RE-F015B	2.55E-02	MISPOSITION OF VALVE F015B
C12-BV_-RE-F021A	1.85E-02	MISPOSITION OF VALVE F021A
P21-BV_-RE-F049A	1.85E-02	MISPOSITION OF RCCW INLET TO CRD HEAT EXCHANGER
P21-BV_-RE-F050A	1.85E-02	MISPOSITION OF RCCW OUTLET FROM CRD HEAT EXCHANGER
C12-BV_-RE-F021B	1.83E-02	MISPOSITION OF VALVE F021B
P21-BV_-RE-F049B	1.83E-02	MISPOSITION OF RCCW INLET TO CRD HEAT EXCHANGER
P21-BV_-RE-F050B	1.83E-02	MISPOSITION OF RCCW OUTLET FROM CRD HEAT EXCHANGER