18 PRA INSIGHTS AFFECTING ESBWR DESIGN

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18 PRA INSIGHTS AFFECTING ESBWR DESIGN

18.1 INTRODUCTION

The purpose of this section is to identify PRA insights and assumptions that significantly affect the outcome of the PRA model. A comprehensive PRA includes not only the quantified results, but also includes information on how the design features affect the risk profile, and knowledge of the uncertainties, assumptions and limitations of the PRA model in representing an estimate of the risks of the plant. Uncertainty, importance, and sensitivity analyses provide important information about areas where certain design features are the most effective in reducing risk with respect to:

- Operation of reactors;
- Hardware failures and human errors;
- Maintaining the "built-in" plant safety and ensuring that the risk does not increase unacceptably;
- Uncertainty associated with the risk estimates; and
- 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 results of the analysis of risk insights and assumptions are presented in Table 18-1. This table is reproduced in DCD Tier 2 Section 19.2. Table 18-2 lists ESBWR design features that contribute to the low core damage frequency and balanced risk profile. In addition, Table 18-3 is a comparison of design features between the ESBWR and currently operating BWRs. The comprehensive analysis of insights and assumptions is presented in Table 18-4.

18.2 PRA ASSUMPTIONS

Uncertainty exists in the PRA model where there is no consensus approach or model and where the choice of approach or model could have an effect on the risk profile such that it influences a decision being made using the PRA. An assumption may be made in response to a source of uncertainty such that a different reasonable alternative assumption would produce different results. During the development phase of the PRA model, aspects of the design that are not fully complete are treated by using modeling assumptions. As such, it is necessary to identify these assumptions and assure that a process exists to maintain their validity. When modeling phenomena or processes in the PRA, different analysts can make different assumptions and still be consistent with the guidelines and standards. The choice of a specific assumption or a particular approximation may, however, influence the results of the PRA. Significant assumptions can be evaluated to determine if the use of alternative methods or numbers could affect the results and insights of the PRA. Sensitivity studies are performed to understand the influence of these differences on risk.

Parameter uncertainties are associated with the values of the fundamental parameters of the PRA model, such as equipment failure rates, initiating event frequencies, and human error probabilities. Model uncertainties reflect aspects of the PRA model where the state of knowledge is incomplete because detailed design information is not available during the design phase of the ESBWR, or different options are available for modeling phenomena (for example, different methods for estimating human error probabilities or common cause failures.) Completeness uncertainty, which is an aspect of model uncertainty, reflects areas of the analysis that are not fully developed. This type of uncertainty is typically treated by using conservative, bounding estimates, or by qualitative conclusions that the item is not risk significant and is screened out from further consideration. Uncertainties are addressed by sensitivity studies in the form of quantified analyses or qualitative arguments with conclusions that provide reasonable assurance that the PRA results represent the as-to-be built plant.

A systematic method is used to identify the significant assumptions of the PRA model. Sections 1.0 through 16.0 of NEDO-33201 are reviewed to identify instances where parametric, modeling and completeness uncertainties have required the use of assumptions. This provides a comprehensive accounting for assumptions from each phase of the PRA development. Assumptions that are clearly conservative and thus would not be considered significant are initially screened out. Each remaining assumption is evaluated as to whether it is made to support a data uncertainty, a design requirement, a modeling preference, or an operator action. This analysis is shown in Table 18-4. With this information and knowledge of the PRA results and importance values from Section 17, a judgment is made as to whether or not the assumption is potentially significant. Such assumptions are further evaluated by a qualitative assessment or sensitivity study. The final result is a list of assumptions that have a significant effect on the PRA model and its insights, and their dispositions. The dispositions are as follows:

- Design Requirement: an assumption that requires specific design details be preserved to maintain its validity.
- Operational Program: an assumption that requires specific operational procedures or training be served to maintain its validity.

- Insight: an assumption that provides significant information about the PRA model or its results that should be maintained in PRA model development, updates, and risk-informed applications

This process does not contain insights about the performance of SSCs, which are captured in the importance analyses that are summarized in Section 17, Table 17.1-2. The list of risk-significant SSCs is used to develop the Reliability Assurance Program, which is discussed in DCD Tier 2 Section 17.4.

18.3 INSIGHTS ON THE ESBWR DESIGN

Insights are considered to be observations or results of the PRA process that should be retained to develop an understanding of its capabilities when applying the model to risk-informed applications. There are insights in the various aspects of the PRA development process. As with assumptions, each earlier section of NEDO-32201 has been reviewed to identify key insights. These insights are also included in Table 18-1.

The advantage of developing the PRA model concurrent with the plant design is in identifying potential risk vulnerabilities and implementing design changes that reduce their risk. In this manner, the risk profile evolves with the design, which lowers the overall risk and reduces dominant risk contributions. The ESBWR PRA has been used extensively during the design phase. Table 18-2 lists design features that contribute to the low core damage frequency and balanced risk profile of the ESBWR. For additional perspective, Table 18-3 provides a comparison of ESBWR vs. BWR design features to illustrate key insights. Review of the proposed design and use of operating experience has led to significant improvements, over currently operating BWR designs, in the plant's ability to prevent and mitigate severe accidents. Significant design improvements include:

The ESBWR front-line safety functions are passive and, therefore, have significantly less reliance on the performance of supporting systems or operator actions. In fact, ESBWR does not require operator actions for successful event mitigation until 72 hours after the onset of an accident.

The ESBWR design reduces the reliance on AC power by using 72-hour batteries for several components. Diesel-driven pumping has been added as a diverse makeup system. The core can be kept covered without any AC sources for the first 72 hours following an initiating fault. This ability significantly reduces the consequences of a loss of preferred (offsite) power initiating fault.

Anticipated Transients Without Scram (ATWS) events are low contributors to plant core damage frequency (CDF) because of the improved scram function and passive boron injection.

The ESBWR design reduces the frequency and consequences of loss of coolant accidents (LOCA) due to large diameter piping by removing the recirculation system altogether.

The design of the ESBWR reduces the possibility of a LOCA outside the containment by designing to the extent practical all piping systems, major system components (pumps and valves), and subsystems connected to the reactor coolant pressure boundary (RCPB) to an ultimate rupture strength at least equal to the full RCPB pressure.

The probability of a loss of containment heat removal is significantly reduced because the Passive Containment Cooling System is highly reliable due to redundant heat exchangers and totally passive component design.

The ESBWR is designed to minimize the effects of direct containment heat, ex-vessel steam explosions, and core-concrete interaction The ESBWR containment is designed to a higher ultimate pressure than conventional BWRs.

The PRA has already been used to identify and quantify various alternatives for improving the reliability of certain design features found in currently operating BWRs. For example, changing the routing of fire suppression piping reduces the probability of internal flooding, which can disable multiple trains of equipment. Following are examples of PRA-based changes that have been incorporated into the ESBWR design, and consequently have contributed to a significant improvement in nuclear safety:

(1) Added redundant, physically separated flow paths to the low pressure injection and suppression pool cooling lines in response to fire analysis.

(2) Determined the loads to be served by the Diverse Protection System, which supplies diverse control signals to safety functions.

(3) Improved the design of digital controls to reduce the likelihood of inadvertent actuation of specified systems.

(4) Added redundant supply valves for Isolation Condenser and Passive Containment Cooling pool makeup.

(5) Added redundant drain line valves for Isolation Condenser System to eliminate a dependency on power supplies.

(6) Changed the routing of fire suppression piping to reduce the likelihood of room flooding.

(7) Determined the appropriate locations of control and instrumentation cabinets and power supplies to ensure physical separation.

(8) Added the Basemat Internal Melt Arrest and coolability device (BiMAC) to reduce the consequences of severe accidents.

Table 18-1

Risk Insights and Assumptions

Insight or Assumption	Disposition
Dominant initiating events for internal events: %T-IORV, %T-GEN, %T-FDW, and %T-LOPP are applied using operating experience data. 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.	Insight
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.	Insight
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).	Insight
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.	Insight
The ESBWR front-line safety functions are passive and, therefore, have significantly less reliance on the performance of supporting systems or operator actions. In fact, ESBWR does not require operator actions for successful event mitigation until 72 hours after the onset of an accident.	Insight
The ESBWR design reduces the reliance on AC power by using 72-hour batteries for several components. Diesel-driven pumping has been added as a diverse makeup system. The core can be kept covered without any AC sources for the first 72 hours following an initiating fault. This ability significantly reduces the consequences of a loss of preferred (offsite) power initiating fault.	Insight
Anticipated Transients Without Scram (ATWS) events are low contributors to plant core damage frequency (CDF) because of the improved scram function and passive boron injection.	Insight
The ESBWR design reduces the frequency and consequences of loss of coolant accidents (LOCA) due to large diameter piping by removing the recirculation system altogether.	Insight

Table 18-1

Risk Insights and Assumptions

Insight or Assumption	Disposition
The design of the ESBWR reduces the possibility of a LOCA outside the containment by designing to the extent practical all piping systems, major system components (pumps and valves), and subsystems connected to the reactor coolant pressure boundary (RCPB) to an ultimate rupture strength at least equal to the full RCPB pressure.	Insight
The probability of a loss of containment heat removal is significantly reduced because the Passive Containment Cooling System is highly reliable due to redundant heat exchangers and totally passive component design.	Insight
The ESBWR is designed to minimize the effects of direct containment heat, ex-vessel steam explosions, and core-concrete interaction The ESBWR containment is designed to a higher ultimate pressure than conventional BWRs.	Insight
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.	Insight
The most significant seismic margins contributor is seismic-induced loss of DC power, and ATWS due to seismic-induced failure of the fuel channels and seismic-induced failure of the SLC tank.	Insight
LOCA frequencies. For each pipe group, the number of lines, the number of sections (assessed on the basis of layout drawings), the frequency apportionments, and the final averaged frequencies. These data are binned into the LOCA initiator classes, as summarized in Section 2, Table 2.3-2. Sensitivity study results indicate that changes in the LOCA frequencies have the potential to impact CDF.	Insight
Sensitivity study results indicate that changes in the human error failure probabilities, particularly pre-initiators, have the potential to impact CDF.	Insight
Sensitivity studyresults indicate that squib valve failure rate estimates have the potential to impact CDF	Insight
Sensitivity study results indicate that changes in test and maintenance unavailability do not significantly impact the CDF or insights.	Insight

Table 18-1

Risk Insights and Assumptions

Insight or Assumption	Disposition
Accident sequences in which DPVs are challenged contribute to approximately 61% of the CDF. In two-thirds of the cases the DPVs are demanded and are successful, and in one-third of the cases the DPVs are demanded and have failed.	Insight
Core damage sequences involving failure of ICS are Class I or III sequences where high pressure makeup has failed and either failure to depressurize occurs or low pressure injection is not available. Given that ICS is failed, the failure of PCCS, or failure to provide make-up to the pools are not significant contributors to core damage frequency.	Insight
The PRA model conservatively assumes that a single failure on either train of SLC caused core damage if the control rods fail to insert (ATWS). The CDF would be reduced by approximately 13% if either train of SLC is able to mitigate ATWS scenarios.	Insight
The results indicate that the pre-initiators have a significant impact on the RAW value. This is primarily due to the large number of potential latent failures, and relatively high reliability for each of these operator actions. The post-initiator HRA screening values are relatively high. As expected, the post-initiator HRA values have a high FV, but a relatively lower impact on RAW.	Insight
An increase of one order of magnitude for vacuum breaker and back-up valve failure rates, only causes the CDF to increases by approximately 10%. However, the Level 2, non-TSL frequency increases by approximately 120% over the baseline results. Steam suppression failures generally lead to core damage states against which release can not be mitigated (class ii-a and class v).	Insight

Table 18-1

Risk Insights and Assumptions

Insight or Assumption	Disposition
Changes to squib valve failure rate, have a significant impact to CDF. As expected, increases in the failure rate of the squib valves used for the ADS cause a significant increase in the accident Class III contribution (core damage at RPV high pressure). Similarly, increases in the failure rate of the squib valves used for the GDCS cause a significant increase in the accident Class I contribution (core damage at RPV low pressure). However, an increase only to the SLC squib valves does not have a very pronounced impact on CDF.	Insight
The ESBWR Level 1 PRA core damage frequency is significantly impacted if the non-safety related systems are not credited. If credit is taken for all the RTNSS systems, the focused Level 1 PRA results are reduced by almost two orders of magnitude. However, the impact to CDF can be minimized, by about one order of magnitude if one only credits the availability of the Diverse Protection System (including surrogate logic for DPS signal for MSIV isolation).	Insight
Including the DPS and ARI functions (ARI function is used as surrogate logic for DPS MSIV isolation that is currently not modeled) with the safety-related systems allows the nTSL release frequency to satisfy the NRC goal of 1E-6/yr for LRF in the internal events, fire, and flooding Level 2 PRA models.	Insight
FAPCS and FPS injection capability provide adequate core cooling for transients given successful DPV or ADS valve operation, even if containment pressure is at the ultimate containment pressure.	Design Requirement
CRD injection is unaffected by containment overpressurization failure. This is an important assumption, based on the containment failure analysis, that supports the use of CRD in these sequences.	Design Requirement

Table 18-1

Risk Insights and Assumptions

Insight or Assumption	Disposition
The DPS cabinet is assumed to be located in a separate fire area in the control building. A preliminary fire PRA analysis model with DPS cabinet located inside room 3301 shows that the fire risk in fire area F3301 would be the dominant contributor to all fire risks due to the high failure probability of common cause failure of software for the safety system, the failure of DPS, and multiple nonsafety-related systems impacted by a fire in room 3301. With a separate fire area for the proposed DPS cabinet in the detailed design, the fire risk can be significantly reduced.	Design Requirement
The ESBWR design features as described in DCD Tier 2 Section 7.1.3 help minimize the adverse affect on safe shutdown due to fire-induced spurious actuations. First of all, the ESBWR instrumentation and control system is digital. A spurious signal cannot be induced by the fire damages in a fiber optic cable. The hard wires are minimized to limit the consequences of a postulated fire. Typically the main control room (MCR) communicates with the safety-related and nonsafety-related DCIS rooms with fiber optics. From the DCIS rooms to the components, fiber optics will also be used up to the Remote Multiplexing Units (RMUs) in the plant. Hard wires then are used to control the subject components. Typically two load drivers are actuated simultaneously in order to actuate the component. To eliminate spurious actuations, these two load drivers are located in different fire areas. Therefore, a fire in a single fire area cannot cause spurious actuation.	Design Requirement
Since the main control room communicates with the DCIS rooms via fiber-optic cables, no spurious actuations due to electrical shorting will be originated from a MCR fire.	Design Requirement
It is assumed that the doors that connect the Control and Reactor Buildings with the Electrical Building galleries are watertight, for flooding of the galleries up to the ground level elevation.	Design Requirement

Table 18-1

Risk Insights and Assumptions

Insight or Assumption	Disposition
The Class IV (ATWS) sequences experience core damage at high pressure because ADS is inhibited as part of the core damage mitigation effort. However, it is assumed that Emergency Operating Procedures (EOPs) will instruct the operator to depressurize after core damage has occurred in an attempt to preserve containment. It is shown in Appendix 8A that the frequency of ATWS sequences experiencing RPV rupture at high pressure is negligible, so only failures at low pressure were analyzed.	Operational Program
Venting is assumed to occur when the containment pressure reaches 90% of the ultimate containment strength.	Operational Program
During shutdown conditions, a fire barrier may not be intact due to maintenance activities. However, an added fire watch would not only increase the success probability of fire detection and suppression, but also help restore the fire barrier in time to prevent fire propagation. Shutdown fire risks related to the fire barriers are evaluated and managed in accordance with the outage risk management program of 10CFR50.65(a)(4).	Operational Program
All LOCAs below TAF during shutdown require closure of lower drywell hatch. The hatch can be opened during shutdown. If a break occurs in the lower drywell and the hatch is not closed, core damage is assumed to occur (once the water level reaches the bottom of the hatch, it is assumed that the door can not be closed and the leak not isolated).	Operational Program
An important recovery action during shutdown is to recover at least one train after loss of both operating RWCU/SDCS trains.	Operational Program
An important recovery action during shutdown is to recover Service Water function after loss of PSW.	Operational Program
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.	Operational Program

Table 18-1

Risk Insights and Assumptions

Insight or Assumption	Disposition
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 hatch(es).	Operational Program
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.	Operational Program
It is assumed that the watertight doors are normally closed at power. Opening of the doors would generate an alarm in the Control Room, and procedures direct their immediate closure upon receipt of an alarm.	Operational Program
It is assumed that, during shutdown, manual and automatic depressurization (ADS) of the vessel are available while the vessel head is in place.	Operational Program
It is assumed that the actuation of the GDCS due to an RPV Level 1 water level signal is available during the entire shutdown period. Improving the reliability of the nitrogen accumulators by providing alarm/indication of accumulator pressure would enhance operator response to low pressure to mitigate the risk of accumulator failures.	Operational Program Operational Program
Improving the reliability of re-filling the pools to maintain long term ICS and PCCS availability would reduce risk of long-term sequences. Actions such as independent verification of valve position would reduce the failure probability.	Operational Program

Table 18-2

ESBWR Design Features That Reduce Risk

Reactor Vessel
Increased volume of water in vessel
No recirculation pump headers minimizes Large Loss-of-Coolant-Accident potential
Smaller diameter piping connected to vessel below core elevation
Isolation Condenser System
Redundant and Diverse active components
Cooling Pools vs. shell-side heat exchangers
In-line condensate reservoirs
Gravity Driven Cooling System
Eliminate reliance on pumps and motor-operated valves
Passive Containment Cooling System
No active components
Independent of AC Power to operate
Standby Liquid Control System
Two pressurized tanks of sodium pentaborate
No pumps required for injection to vessel
Reactor Water Cleanup/Shutdown Cooling
Uses larger heat exchangers for backup decay heat removal
Full pressure shutdown cooling capability
Fuel and Auxiliary Pool Cooling System
Low Pressure Coolant Injection mode for backup coolant injection
Automatic Suppression Pool Cooling mode
Control Rod Drive System
Provides high pressure, high capacity injection to vessel
ATWS Prevention/Mitigation
Scram Discharge Volume eliminated
Fine Motion Control Rod Drives provide diverse backup
Automatic, safety-related Standby Liquid Control System
Alternate Rod Insertion

Table 18-2

ESBWR Design Features That Reduce Risk

Instrumentation and Control
Multiple diverse systems to minimize common cause failures
Severe Accident Mitigation
BiMAC device added to eliminate the uncertainty of ex-vessel debris coolability and core-concrete interaction gas generation
Fire water injection capable of arresting core melt in-vessel (not modeled in PSA)
Inert containment prevents hydrogen combustion
High ultimate rupture strength of containment
Loss of Preferred Power
Plant capable of “island mode” of operation in the event of loss of grid (not modeled in PSA)

Table 18-3

Comparison of BWR vs. ESBWR PRA Prevention and Mitigation Functions

Prevention or Mitigation Function	**BWR Features**	**ESBWR Features**	**Net Effect of ESBWR Design and Operation Features**
Intiating Events - Transients	Turbine Trip, Loss of Offsite Power, Loss of FW	Turbine Trip, Loss of Offsite Power, Loss of FW	Similar
Initiating Events – LOCAs, Line Breaks Outside Cont., ISLOCA, Reactor Vessel Rupture	Small, Medium, Large LOCA, LBOC, ISLOCA, Vessel Rupture	Small, Medium, Large LOCA, LBOC, ISLOCA, Vessel Rupture	ESBWR has significantly less large bore piping outside of vessel due to elimination of Recirculation Pumps. Large LOCA and ISLOCA frequencies are lower. ESBWR high to low pressure interfaces use piping capable of withstanding vessel rupture pressure – reduces Interfacing Systems LOCAs.
Reactivity Control	RPS ARI RPT SLC ATWS RPV Level Control	RPS ARI FW Runback SLC ATWS RPV Level Control	Similar Similar ESBWR FW Runback performs similar function to BWR RPT. BWR SLC requires AC Power, ESBWR SLC is accumulator-driven and does not require AC Power to initiate. Level Control treated similarly. ESBWR ADS Inhibit function is automatic.
High Pressure Mitigation	HPCS/ HPCI RCIC CRD	Isolation Condenser CRD 4-Division Digital	ESBWR does not have a high pressure coolant injection pump. CRD pump capacity and discharge head are enhanced in ESBWR to provide high pressure injection capability. ESBWR uses Isolation Condenser to mitigate transients and prevent need to depressurize. Isolation Condenser does not require AC or DC power, or operator action to control vessel level by controlling pump flow.

Table 18-3

Comparison of BWR vs. ESBWR PRA Prevention and Mitigation Functions

Prevention or Mitigation Function	BWR Features	ESBWR Features	Net Effect of ESBWR Design and Operation Features
	2-Division Analog Actuation	Actuation	
Depressurization	Manual SRVs Automatic ADS SRVs	Manual SRVs Automatic ADS SRVs ADS DPVs	ESBWR uses SRVs and DPVs. DPVs are squib-actuated and do not reclose. ESBWR uses manual depressurization with SRVs to preclude the need for ADS. No re-pressurization with DPVs
Low Pressure Mitigation	LPCS LPCI Fire Water Injection	GDCS Injection GDCS Equalize FAPCS LPCI Fire Water Injection	Both GDCS subsystems are independent of AC or DC power. FAPCS LPCI function provides injection with in-line heat exchanger.
Containment Heat Removal	RHR Heat Exchangers Venting	PCCS FAPCS Supp. Pool Cooling RWCU SDC Venting	PCCS is independent of AC or DC power. RWCU SDC provides high pressure cooling.
Supporting Functions	AC Distribution Diesel Generators DC Distribution Component Cooling Room Cooling	AC Distribution Diesel Generators DC Distribution Component Cooling	ESBWR Passive Safety-Related Systems require no Supporting Functions for 72 hours. ESBWR DC System uses 72 hour capacity batteries for safety-related functions. BWR vs. ESBWR component and room cooling are similar.

Table 18-3

Comparison of BWR vs. ESBWR PRA Prevention and Mitigation Functions

Prevention or Mitigation Function	BWR Features	ESBWR Features	Net Effect of ESBWR Design and Operation Features
		Room Cooling	
Instrumentation and Control Systems	Analog single-failure proof. Digital in limited use (FW level controller in some plants.)	Digital Controls. Diverse Protection System.	Digital Controls with triple-redundancy. Diverse Control for key functions to eliminate the effects on common-cause failures, e.g., software.
Severe Accident Mitigation	Severe Accident Guidelines	Severe Accident Guidelines BiMAC	BiMAC reduces the containment failure probability from core-concrete interaction.

Table 18-4

Analysis of Insights and Assumptions

Insight or Assumption	Screening Basis	Potentially Significant?	Disposition
SPC & LPCI success. The correct operation of one of the pumps, with its corresponding heat exchanger, is necessary in both SPC and LPCI operating modes, along with the proper alignment of the corresponding intake and discharge.	Modeling preference	Yes	Sensitivity study changed the success criteria to 2 pumps - Not Significant
The Class IV (ATWS) sequences experience core damage at high pressure because ADS is inhibited as part of the core damage mitigation effort. However, it is assumed that Emergency Operating Procedures (EOPs) will instruct the operator to depressurize after core damage has occurred in an attempt to preserve containment. It is shown in Appendix 8A that the frequency of ATWS sequences experiencing RPV rupture at high pressure is negligible, so only failures at low pressure were analyzed.	Modeling preference	Yes	Operational Program
Venting is assumed to occur when the containment pressure reaches 90% of the ultimate containment strength.	Modeling preference	Yes	Operational Program
During shutdown conditions, a fire barrier may not be intact due to maintenance activities. However, an added fire watch would not only increase the success probability of fire detection and suppression, but also help restore the fire barrier in time to prevent fire propagation. Shutdown fire risks related to the fire barriers are evaluated and managed in accordance with the outage risk management program of 10CFR50.65(a)(4).	Modeling preference	Yes	Operational Program

Table 18-4

Analysis of Insights and Assumptions

Insight or Assumption	Screening Basis	Potentially Significant?	Disposition
All LOCAs below TAF during shutdown require closure of lower drywell hatch. The hatch can be opened during shutdown. If a break occurs in the lower drywell and the hatch is not closed, core damage is assumed to occur (once the water level reaches the bottom of the hatch, it is assumed that the door can not be closed and the leak not isolated).	Modeling preference	Yes	Operational Program
An important recovery action during shutdown is to recover at least one train after loss of both operating RWCU/SDCS trains.	Modeling preference	Yes	Operational Program
An important recovery action during shutdown is to recover Service Water function after loss of PSW.	Modeling preference	Yes	Operational Program
ADS and relief-only SRVs will be significantly different. This assumption allows the two different types of SRVs to be in separate common cause failure groups. Assumption was based on the different functions and on discussion with the system engineer pertaining to the future detailed design of the valves.	Modeling preference	Yes	Not significant per SRV/CCF sensitivity study.
Bypassing an APRM channel for testing is assumed to occur weekly and take 1/2 hour.Unavailability values is estimate as 0.003 = (26/8760)	Data	Yes	Not significant due to low RAW value

Table 18-4

Analysis of Insights and Assumptions

Insight or Assumption	Screening Basis	Potentially Significant?	Disposition
Four of twelve turbine bypass valves must open for turbine bypass function to be successful for PCS. Since one of four feedwater pumps and one of four condensate pumps must be successful and each feedwater and condensate pump is 33% capacity, the capacity of feedwater/condensate going into the reactor is about 33%.	Data	Yes	Not significant based on sensitivity study
Although fire areas F3301 and F3302 are well separated by the corridor F3100, it is conservatively assumed that fire propagation could occur via two pairs of fire doors with a fire barrier failure probability of 2*7.4E-3*7.4E-3 = 1.1E-4.	Data	Yes	Not significant based on fire barriers sensitivity study
Similarly, although RWCU pump rooms are well separated, it is conservatively assumed that a fire propagation could occur via the fire door at elevation 4650 between fire areas F1152 and F1162, which could fail the controls to both RWCU trains. The fire barrier failure probability for this case is then assumed to be 7.4E-3.	Data	Yes	Not significant based on fire barriers sensitivity study
Losses of both RCCWS trains, both instrument air trains and both PIP buses due to fire propagation are also assumed to result in loss of RWCU. Fire areas F4250 & F4260, and F4350 & F4360 are separated by walls. Thus a fire barrier failure rate of 1.2E-3 is assumed. Fire areas F5550 & F5560 are separated by a corridor (fire area F5100). It is conservatively assumed that fire propagation could occur via three pairs of fire doors with a fire barrier failure probability of 3*7.4E-3*7.4E-3 = 1.6E-4.	Data	Yes	Not significant based on fire barriers sensitivity study

Table 18-4

Analysis of Insights and Assumptions

Insight or Assumption	Screening Basis	Potentially Significant?	Disposition
The fire areas in EB are well separated. Fire propagation from one area to another in EB won't cause LOPP until it propagates to a third area. Fire propagation between the two cable tunnels could result in a scenario similar to LOPP. Since the fire barriers between the two tunnels are walls and sealed penetrations, a fire barrier failure probability of 1.2E-3 is used. A fire in switchyard is conservatively assumed to result in a loss of preferred offsite power.	Data	Yes	Not significant based on fire barriers sensitivity study
Mission Time The design of the ESBWR is such that the onsite inventory of cooling water available and plant battery capacity can keep the core covered using passive systems for more than 72 hours. However, the simplifying assumptions made in the PRA analysis are not always applicable for mission times longer than 24 hours. For example, the PRA assumes that once the initiator has occurred, no credit will be given for repair of failed equipment. This is a conservative assumption that is reasonable for a mission time of 24 hours; it provides unreasonable results and misleading insights for a 72 hour mission. Therefore, the mission time for the ESBWR PRA is 24 hours. A sensitivity analysis in Section 11 characterizes the effects of extending mission times to 72 hours and concludes that it is not significant.	Modeling preference	Yes	Not significant - sensitivity study on mission times.

Table 18-4

Analysis of Insights and Assumptions

Insight or Assumption	Screening Basis	Potentially Significant?	Disposition
Because a majority of the ICS piping at RPV pressure is inside containment, the percentage of ICS piping outside containment is estimated to be 10%. An estimate of the frequency for this initiating event is 10% of the LOCA frequency apportioned for IC outside the containment in Table 2.3- 1.and is 10% of the sum of DPV line/IC (7.55E-06/yr) and IC return lines (7.66E-06/yr) or 1.53E-06.	Data	Yes	Not Significant - Sensitivity Study Assumed 50% of ICS piping outside of containment
LOCA frequencies. For each pipe group, the number of lines, the number of sections (assessed on the basis of layout drawings), the frequency apportionments, and the final averaged frequencies. These data are binned into the LOCA initiator classes, as summarized in Section 2, Table 2.3-2.	Data	Yes	Insight
The ESBWR front-line safety functions are passive and, therefore, have significantly less reliance on the performance of supporting systems or operator actions. In fact, ESBWR does not require operator actions for successful event mitigation until 72 hours after the onset of an accident.	Results	Yes	Insight
The ESBWR design reduces the reliance on AC power by using 72-hour batteries for several components. Diesel-driven pumping has been added as a diverse makeup system. The core can be kept covered without any AC sources for the first 72 hours following an initiating fault. This ability significantly reduces the consequences of a loss of preferred (offsite) power initiating fault.	Results	Yes	Insight

Table 18-4

Analysis of Insights and Assumptions

Insight or Assumption	Screening Basis	Potentially Significant?	Disposition
Anticipated Transients Without Scram (ATWS) events are low contributors to plant core damage frequency (CDF) because of the improved scram function and passive boron injection.	Results	Yes	Insight
The ESBWR design reduces the frequency and consequences of loss of coolant accidents (LOCA) due to large diameter piping by removing the recirculation system altogether.	Results	Yes	Insight
The design of the ESBWR reduces the possibility of a LOCA outside the containment by designing to the extent practical all piping systems, major system components (pumps and valves), and subsystems connected to the reactor coolant pressure boundary (RCPB) to an ultimate rupture strength at least equal to the full RCPB pressure.	Results	Yes	Insight
The probability of a loss of containment heat removal is significantly reduced because the Passive Containment Cooling System is highly reliable due to redundant heat exchangers and totally passive component design.	Results	Yes	Insight
The ESBWR is designed to minimize the effects of direct containment heat, ex-vessel steam explosions, and core-concrete interaction The ESBWR containment is designed to a higher ultimate pressure than conventional BWRs.	Results	Yes	Insight
Dominant initiating events for internal events: %T-IORV conservative treatment; %T-GEN, %T-FDW, and %T-LOPP are applied using operating experience data. 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.	Results	Yes	Insight

Table 18-4

Analysis of Insights and Assumptions

Insight or Assumption	Screening Basis	Potentially Significant?	Disposition
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.	Results	Yes	Insight
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).	Results	Yes	Insight
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.	Results	Yes	Insight
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.	Results	Yes	Insight
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.	Results	Yes	Operational Program

Table 18-4

Analysis of Insights and Assumptions

Insight or Assumption	Screening Basis	Potentially Significant?	Disposition
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 hatch(es).	Results	Yes	Operational Program
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.	Results	Yes	Operational Program
The most significant seismic margins contributor is seismic-induced loss of DC power, and ATWS due to seismic-induced failure of the fuel channels and seismic-induced failure of the SLC tank.	Results	Yes	Insight
Sensitivity study results indicate that changes in the human error failure probabilities, particularly pre-initiators, have the potential to impact CDF.	Results	Yes	Insight
Sensitivity studyresults indicate that squib valve failure rate estimates have the potential to impact CDF	Results	Yes	Insight
Sensitivity study results indicate that changes in test and maintenance unavailability do not significantly impact the CDF or insights.	Results	Yes	Insight
Sensitivity study results indicate that changes in the LOCA frequencies have the potential to impact CDF.	Results	Yes	Insight

Table 18-4

Analysis of Insights and Assumptions

Insight or Assumption	Screening Basis	Potentially Significant?	Disposition
FAPCS and FPS injection capability provide adequate core cooling for transients given successful DPV or ADS valve operation, even if containment pressure is at the ultimate containment pressure.	Modeling preference	Yes	Design Requirement
CRD injection is unaffected by containment overpressurization failure. This is an important assumption, based on the containment failure analysis, that supports the use of CRD in these sequences.	Modeling preference	Yes	Design Requirement
The DPS cabinet is assumed to be located in a separate fire area in the control building. A preliminary fire PRA analysis model with DPS cabinet located inside room 3301 shows that the fire risk in fire area F3301 would be the dominant contributor to all fire risks due to the high failure probability of common cause failure of software for the safety system, the failure of DPS, and multiple nonsafety-related systems impacted by a fire in room 3301. With a separate fire area for the proposed DPS cabinet in the detailed design, the fire risk can be significantly reduced.	Modeling preference	Yes	Design Requirement

Table 18-4

Analysis of Insights and Assumptions

Insight or Assumption	Screening Basis	Potentially Significant?	Disposition
The ESBWR design features as described in DCD Tier 2 Section 7.1.3 help minimize the adverse affect on safe shutdown due to fire-induced spurious actuations. First of all, the ESBWR instrumentation and control system is digital. A spurious signal cannot be induced by the fire damages in a fiber optic cable. The hard wires are minimized to limit the consequences of a postulated fire. Typically the main control room (MCR) communicates with the safety-related and nonsafety-related DCIS rooms with fiber optics. From the DCIS rooms to the components, fiber optics will also be used up to the Remote Multiplexing Units (RMUs) in the plant. Hard wires then are used to control the subject components. Typically two load drivers are actuated simultaneously in order to actuate the component. To eliminate spurious actuations, these two load drivers are located in different fire areas. Therefore, a fire in a single fire area cannot cause spurious actuation.	Modeling preference	Yes	Design Requirement
Since the main control room communicates with the DCIS rooms via fiber-optic cables, no spurious actuations due to electrical shorting will be originated from a MCR fire.	Modeling preference	Yes	Design Requirement
It is assumed that the doors that connect the Control and Reactor Buildings with the Electrical Building galleries are watertight, for flooding of the galleries up to the ground level elevation.	Modeling preference	Yes	Design Requirement
It is assumed that the watertight doors are normally closed at power. Opening of the doors would generate an alarm in the Control Room, and procedures direct their immediate closure upon receipt of an alarm.	Modeling preference	Yes	Operational Program
It is assumed that, during shutdown, manual and automatic depressurization (ADS) of the vessel are available while the vessel head is in place.	Modeling preference	Yes	Operational Program

Table 18-4

Analysis of Insights and Assumptions

Insight or Assumption	Screening Basis	Potentially Significant?	Disposition
It is assumed that the actuation of the GDCS due to an RPV Level 1 water level signal is available during the entire shutdown period.	Modeling preference	Yes	Operational Program
Accident sequences in which DPVs are challenged contribute to approximately 61% of the CDF. In two-thirds of the cases the DPVs are demanded and are successful, and in one-third of the cases the DPVs are demanded and have failed.	Results	Yes	Insight
Core damage sequences involving failure of ICS are Class I or III sequences where high pressure makeup has failed and either failure to depressurize occurs or low pressure injection is not available. Given that ICS is failed, the failure of PCCS, or failure to provide make-up to the pools are not significant contributors to core damage frequency.	Results	Yes	Insight
The analysis does not take credit for any fire suppression (i.e., self-extinguishment, installed suppression systems, nor manual fire fighting activities).	Modeling preference	No	
The analysis assumes that all fires disable all potentially affected equipment in the area.	Modeling preference	No	
The analysis assumes that all fire-induced equipment damage occurs at t=0.	Modeling preference	No	
The dominant failure mode in ATWS sequences is an assumed failure of the control rods to insert into the core due to mechanical binding.	Results	No	
A fire in the control room does not affect the automatic actuations of the safety systems.	Results	No	

Table 18-4

Analysis of Insights and Assumptions

Insight or Assumption	Screening Basis	Potentially Significant?	Disposition
Remote shutdown panels allows the opportunity to perform manual actuations for failed automatic actuations that may occur.	Results	No	
No latent error has been modeled for the ICS pool subcompartment isolation valves. These valves are controlled by tech specs and surveillances. It is assumed that there will be sufficient controls and/or remote position indication to warrant not modeling latent mispositioning errors.	Modeling preference	No	
A detailed design does not exist for the NDCIS system. This model was built using input from the I&C design group.	Conservative assumption on design	No	
RPS uses separate and diverse components than the ESF I&C systems, and NMS. The diverse components include sensors, software and hardware. These systems are currently being designed and are expected to be diverse. It is expected that NMS system are also diverse from RPS. No common causes are included between these systems even if components perform similar functions TLU, OLU, DTM, etc.)	Conservative assumption on design	No	
It is assumed that the backup nitrogen bottle racks are able to supply nitrogen to the systems supported during accident sequences at required pressure for 24 or 72 hours.	Modeling preference	no	
ICS auxiliary line is not required for success.The system has a built-in auxiliary flowpath that takes steam from the heat exchanger inlet and returns it to the main steam line. The purpose is to ensure that the supply line to the heat exchanger is always purged. It is assumed that this line and F013 are not required for success.	Modeling preference	No	

Table 18-4

Analysis of Insights and Assumptions

Insight or Assumption	Screening Basis	Potentially Significant?	Disposition
%BOC-IC is the initiating event Break Outside Containment - Isolation Condenser. The break is assumed to be on loop 1(A).	Modeling preference	No	
Due to the inherent simplicity of ICS (i.e., no active components), maintenance unavailability has not been modeled. Tech specs do allow system unavailability, but that has not been modeled.	Modeling preference	No	
The testing described in the ICS system description does not inop the system. The components that are aligned during the test automatically realign to their normal position on actuation of the system.	Modeling preference	No	
ICS has provisions for venting of non-condensable gasses from the upper and lower headers of the heat exchanger. These are required only if gasses accumulate. It is assumed that they will.	Conservative assumption on design	No	
The failure of the common IC/PCC pools due to catastrophic rupture is not considered in the model. The only plausible failure mechanism is due to external events, such as a seismic event.	Modeling preference	No	
The PANTHER report shows that gas accumulation at the top of the IC heat exchangers does not contribute to loss of heat removal capability. It also demonstrates that the top vents are not effective at removing gasses (ie, the removal of gasses doesn't change overall heat removal capability). Therefore, the vents are not modelled.	Modeling preference	No	

Table 18-4

Analysis of Insights and Assumptions

Insight or Assumption	Screening Basis	Potentially Significant?	Disposition
The ICS valve types for F104A, B, C and D are not yet specified. It is assumed that they will be a combination of NOV and NMOV, with one of each type feeding each of the two pools. Valves are assumed to fail as-is. Assume that there will be nitrogen accumulators.	Conservative assumption on design	No	
The CRD pumps are tripped to terminate CRD system flow on receipt of low water level signals from two of the three GDCS pools (when in high pressure makeup mode of operation). It would then seem the failure of those signals would prohibit start of the pumps.	Conservative assumption on design	No	
SLCS Valves F030A&B constantly monitored. No failure event modeled due to constant monitoring.	Modeling preference	No	
SLCS valves F507A&B and F508A&B SOVs fail closed and opened from MCR. These SOVs are not modeled because they are constantly monitored.	Modeling preference	No	
SLCS squib firing signal and power come from C63 and C72. The RMUs are still in design; however, the current thinking is that a power supply will be added to the RMU to supply the required DC Voltage to fire the squibs as well as the initiation signal.	Conservative assumption on design	No	
Suppression pool temperature elements not modeled in detail.Each suppression pool has 12 temperature elements. The readings from each of these sensors is averaged, with the failed ones and ones not submerged excluded. As such, the hardware portion of this signal is extremely reliable.	Modeling preference	No	

Table 18-4

Analysis of Insights and Assumptions

Insight or Assumption	Screening Basis	Potentially Significant?	Disposition
Backup scram signal to come from primary RPS OLU. No information is available for the design of the backup scram. It has been assumed that two relays in series are required to close to energize the solenoid for air dump valve F036. The first relay is assumed to close on input from the primary RPS Div I	Conservative assumption on design	No	
Bypass of a TLU is assumed to occur during testing of the corresponding channel of instrumentation.Channel testing is expected to test all the necessary components for the generation of the signal.	Modeling preference	No	
Include APRM channel test as a failure (false signal)Bypassing an APRM channel generates a trip signal for a specific division of RPS (2 of 4 NMS logic). This is conservative, but the test is included since the logic adjust to 2 of 3 trip signal for RPS when a single channel is bypassed.	Modeling preference	No	
Manual scram. This operator action causes de-energizing of the series-parallel circuits of load drivers, by opening switches downstream of the RPS logic, which provide power to the scram pilot valve solenoids. This is treated as recovery action if necessary.	Modeling preference	No	
A SPECIAL EVENT was developed to address (across all DPS actuation) COMMON CAUSE FAILURE OF DPS LOAD DRIVERS. The CCFspecial event was named C72-LDD-CF-LOADS and assigned a probability equal to that CCF of RPS load drivers in 23A6100 ABWR SAFETY ANALYSIS REPORT	data	No	
ADS uses L1 signal to actuate DPVs. The same L1 signals (channels) from NBS are supplied to each individual DPS processor via analog trip units. The signal is transferred via C62.	Conservative assumption on design	No	

Table 18-4

Analysis of Insights and Assumptions

Insight or Assumption	Screening Basis	Potentially Significant?	Disposition
Manual actuation of the the SRVs is the preferred method to depressurize since it prevents a DPV steam release to the containment. Since the event trees allow this operator action prior to ADS, failure of manual SRV actuation is assumed to only actuate DPVs.	Modeling preference	No	
Assume each ATWS Logic Processor is powered by a separate safety related divisional power supply in the control building. The following power supplies are assumed: R13-11-CB-ST, R13-21-CB-ST, R13-31-CB-ST, and R13-41-CB-ST.	Conservative assumption on design	No	
Assume each DPS processor is powered by a separate non-divisional power supply in control building. The following supplies are used R13-NSR-CBA-ST, R13-NSR-CBB-ST, R13-NSR-CBC-ST	Conservative assumption on design	No	
Assume DPS fails to initiate ICS on L 2. The same L2 signals (channels) from NBS are supplied to each individual DPS processor via analog trip units. The signal is transferred via C62.	Modeling preference	No	
It is assumed that the feedwater run back signal is provided to two load drivers in series to generate a runback in the feedwater system.Feedwater runback uses the same actuation high flux (SRNM) signals used by SLC. The combination of high flux with high RPV dome pressure signal is used to actuate FW runback. The signal is further transferred to DPS for additional processing.	Conservative assumption on design	No	

Table 18-4

Analysis of Insights and Assumptions

Insight or Assumption	Screening Basis	Potentially Significant?	Disposition
It is assumed that the high pressure combined with high flux signals (SRNM) generates all non-LOCA actuations. The low-level combined with high flux actuation is used only for LOCA initiating events.SLC actuation is based on DCD Figure 7.8-3. The SLC actuation signals are processed by the ATWS logic processors (SSLC) for each division. SLC uses high flux combined with either L2, or high-pressure actuation signals.	Conservative assumption on design	No	
Loss of power to two of the three electrical FMCRD groups is assumed to result in failure of the FMCRD Run In function for ATWS mitigation.	Conservative assumption on design	No	
The power supply for the feedwater run back circuit is assumed to be triply redundant.POWER FAILURE FOR FW RUNBACK CKT requires failure of the three non safety related UPS turbine building groups. The following non-safety related supplies are modeled: R13-NSR-TBA-ST, R13-NSR-TBB-ST, R13-NSR-TBC-STR	Conservative assumption on design	No	
The same L2 signal sensors are used by DPS for both ARI, FMCRD "Run In", and ICS actuations	Conservative assumption on design	No	
There is no common cause between APRM and SRNM components.Systems are diverse	Modeling preference	No	
GDCS Injection Line Check Valves might not be of standard design. The specific design of the check valves in the injection path is not yet known. A conservatism of a factor of 10 with respect to the typical check valve/application is utilized in the model until more certainty can be determined.	Conservative assumption on design	No	

Table 18-4

Analysis of Insights and Assumptions

Insight or Assumption	Screening Basis	Potentially Significant?	Disposition
No CCF between GDCS Injection and Equalizing Squib Valves necessary. There will be sufficient difference in the design or manufacture between the squib valves used in the injection lines from those used in the equalizing lines to minimize potential common cause failures between them.	Modeling preference	No	
FAPCS automatically actuates in the suppression pool cooling (SPC) mode on a suppression pool high temperature signal, whenever the system has not been previously aligned for low pressure coolant injection mode. This is modeled for the standby train	Modeling preference	No	
FAPCS is assumed to fail (in both modes modeled) if the MOVs to the bypass lines fail to open. For SPC and LPCI, the filter/demin section of the system is isolated, and flow bypasses the cleaning funtion in the system. Valves F013A and B open.	Modeling preference	No	
No additional manual valves have been included in the FAPCS model that could be closed to isolate an individual pump or heat exchanger for maintenance. This is due to the constant running of the system, the monthly rotation of the trains, and the ability to isolate.	Modeling preference	No	
FAPCS flow diversion or inadvertent suction (from a non preferred source) that require two failures have not been modeled. These scenarios generally involve two passive failures (fail to remain closed), and are much less likely than the single active failures	Modeling preference	No	
FAPCS function is assumed lost (in both modeled modes) if the return MOVs to the Spent Fuel & Auxiliary pools fail to close or remain closed. F008A/B must close to isolate FAPCS flow from diverting from the desired (SPC or LPCI) path to the other pools.	Modeling preference	No	

Table 18-4

Analysis of Insights and Assumptions

Insight or Assumption	Screening Basis	Potentially Significant?	Disposition
Several signals can trip the FAPCS pumps. A generic event of 'spurious pump trip' is included to account for several of the one that are not explicitly modeled. The ones that are explicitly modeled (pump flow transmitter, suction pressure transmitter	Conservative assumption on design	No	
FAPCS is assumed to fail if the return MOVs from the spent & auxiliary pools fail to close (or remain closed). It would not cause loss of system function, but operators would isolate the system if the spent fuel pool level was dropping	Conservative assumption on design	No	
Design of the suppression pool suction valves is still being developed. Current design documents have the valves as Air Operated Valves (ACV). System engineers claim an update is in process to make the valves Nitrogen Operated Valves (NOV).	Modeling preference	No	
Components in the FAPCS system are assumed to have either a monthly test interval, or a two year test interval. Equipment in line with the normal operation of the system is rotated monthly.	Conservative assumption on design	No	
1/2 of RWCU/SDC LLOCA Steam Breaks are in Train A, 1/2 are in Train B.There is an equal probability that the large LOCA will occur on Train A or Train B.	Modeling preference	No	
50% of the time RWCU/SDC Train A will be running, 50% of the time Train B will be running. The system is in operation during all modes of reactor operation. Because the system consists of two redundant trains, only one train is needed at power	Modeling preference	No	

Table 18-4

Analysis of Insights and Assumptions

Insight or Assumption	Screening Basis	Potentially Significant?	Disposition
Common cause was not modeled for passive failures where the active failures dominated. For example, if the failure of an MOV to open is 4.00E-03 and the failure to remain open for 24 hours is 3.36E-06, common cause will be modeled for the valve to open (the active failure) but not for the failure of the valve to remain open for 24 hours	Modeling preference	No	
Supply to RWCU/SDC pump suction will require both suction from vessel bottom and mid vessel connections.	Modeling preference	No	
Common cause failure for condensate pumps, feedwater pumps and booster feedwater pumps fail to start is not modeled.Because 3 of 4 pumps are normally running, only one pump is required to start. Therefore, common cause failure to start does not apply to the feedwater pumps, feedwater booster pumps or condensate pumps.	Modeling preference	No	
Common cause was not modeled for passive failures where the active failures dominated.For example, if the failure of an MOV to open is 4.00E-03 and failure to remain open for 24 hours is 3.36E-06, common cause will be modeled for the valve to open (the active failure) but not for the failure of the valve to remain open for 24 hours (the pasive failure)	Modeling preference	No	
Spurious signal to close MSIV is not modeled.To fail the main steam function four spurious signals would be required. The probability of four spurious signals occurring within the 24 hour mission time is a rare event and will not be considered a credible failure of the main steam function.	Modeling preference	No	

Table 18-4

Analysis of Insights and Assumptions

Insight or Assumption	Screening Basis	Potentially Significant?	Disposition
Standby feedwater pump will auto start if one of the running pumps trip.The Cond&FDWS shall be designed to start the standby feedwater pump automatically through the ASD if one of the operating pump trips	Conservative assumption on design	No	
The assumed arrangement of the Circulating Water System is all four pumps running. During normal power operations, three or four pumps will be operating. This assumpion slightly simplifies the modeling and has minimal overal impact on the model results.	Conservative assumption on design	No	
Each of the four pump trains in the system is assumed to be out of service for maintenance one week per year.	Modeling preference	No	
No valves are included in the system model for the CIRC system. The failure of the valves associated with the pumps operating during normal operation is not modeled because multiple failures are required to fail the function.	Modeling preference	No	
Loss of control signal will cause RCCW HX Bypass Valves (F016A, F016B) to Fail Closed. Because the valves are fail closed, it is not necessary to model loss of control signal or loss of power to the control signal. The function of the system is to provide cooling to the loads.	Modeling preference	No	
Loss of control signal will cause RCCW HX Flow Control Valves (F012A, F012B) to Fail Open. Because the valves are fail open valves, it is not necessary to model loss of control signal or loss of power to the control signal.	Modeling preference	No	

Table 18-4

Analysis of Insights and Assumptions

Insight or Assumption	Screening Basis	Potentially Significant?	Disposition
Loss of flow due to pipe breaks, open drain valves or relief valves is not modeled.Loss of flow due to pipe breaks or open drain valves is not included in this analysis due to the low probability of occurrence. In addition, loss of flow through relief valves is not judged to be significant flow diversion and is not modeled.	Modeling preference	No	
Maintenance unavailability of PSW valves associated with an RCCW HX are included with the HX unavailability.For example, maintenance on PSW system valves P41-F006A or P41-F0007A would remove HX-0001A3 from service. This out of service time caused by the maintenance of the PSW supply and return valves to the heat exchanger would be included with the heat exchanger.	Modeling preference	No	
Motor operated valves on the service water discharge are normally open and air operated valves on the service water supply are normally closed when HX is not in service.It is assumed that the AOV on the PSW inlet to the RCCW HX is normally closed and is used to isolate the heat exchanger.	Modeling preference	No	
Loss of control signal or instrument air will cause the TCCW HX Bypass Valve (F0008) to Fail Closed.The TCCW HX Bypass Valve (F0008) is a fail closed valve. Because the valve is a fail closed valve, it is not necessary to model loss of control signal, loss of power to the control signal or loss of instrument air.	Modeling preference	No	
Loss of control signal or instrument air will cause the TCCW HX Flow Control Valve (F0006) to Fail Open.The TCCW HX Flow Control Valve (F0006) is a fail open valve. Because the valve is a fail open valve, it is not necessary to model loss of control signal, loss of power to the control signal or loss of instrument air.	Modeling preference	No	

Table 18-4

Analysis of Insights and Assumptions

Insight or Assumption	Screening Basis	Potentially Significant?	Disposition
Maintenance unavailability of PSW valves associated with an TCCW HX are included with the HX unavailability.For example, maintenance on PSW system valves P41-F001A or P41-F0008A would remove HX-0001A from service. This out of service time caused by the maintenance of the PSW supply and return valves to the heat exchanger would be included with the heat exchanger.	Modeling preference	No	
PSW operating pumps and heat exchangers will be rotated on a quarterly basis.While it is expected that the operating pumps and heat exchangers will be rotated to standby basis more often, it is assumed that they will be rotated quarterly.	Conservative assumption on design	No	
PSW Discharge Isolation MOVs from the TCCW HXs are normally open. PSW Supply Isolation AOVs to the TCCW HXs are normally closed.It is assumed that the AOV on the PSW inlet to the RCCW HX is normally closed and is used to isolate the heat exchanger. Because the MOVs are normally open and the desired position is open, no power or control dependencies are modeled for the PSW Discharge MOVs.	Modeling preference	No	
The heat exchangers are operated in each configuration one sixth of the time.Two of the four heat exchangers are in service during normal operation. There are six possible configurations for two heat exchangers to be in service. The six possible combinations are 1A-1B, 1A-1C, 1A-1D, 1B-1C, 1B-1D, 1C-1D.	Modeling preference	No	
Two of the three TCCW pumps are in service during normal operation. There are three possible configurations for two pumps to be in service. The three possible combinations are 1A-1B, 1A-1C, 1B-1C. It is assumed that there is no preferred configuration	Modeling preference	No	

Table 18-4

Analysis of Insights and Assumptions

Insight or Assumption	Screening Basis	Potentially Significant?	Disposition
Unavailability due to Testing or Maintenance for the TCCW Heat Exchangers is 2.2E-03.Unavailability due to Testing or Maintenance for the TCCW heat exchangers is assumed to be the same as the emergency service water system in EPRI ALWR plus 10%.	Data	No	
Unavailability due to Testing or Maintenance for the TCCW pumps is 2.00E-3/year.Unavailability due to Testing or Maintenance for the TCCW pumps is assumed to be the same as the emergency service water system in EPRI ALWR.	Data	No	
Vent lines, drain lines, instrument lines and test lines have not been included in the system model.Because these lines are small in diameter, may have manual isolation valves and/or are rotated into service on a periodic basis which would identify mispositioning errors, they are not included in the system model.	Modeling preference	No	
Ventilation is not necessary for TCCW since the pumps and HXs are located in the open area of the turbine building.The TCCW pumps and HXs are located on the bottom elevation of the turbine building and the area is open to the turbine building.	Conservative assumption on design	No	
Common cause was not modeled for passive failures where the active failures dominate.For example, if the failure of an MOV to open is 4.00E-03 and failure to remain open for 24 hours is 3.36E-06, common cause will be modeled for the valve to open (the active failure) but not for the failure of the valve to remain open for 24 hours (the passive failure.)	Modeling preference	No	
Cross connect valve between cooling tower A and cooling tower B is normally open.With the cross connect normally open, either cooling tower can be used to cool either PSW train.	Conservative assumption on design	No	

Table 18-4

Analysis of Insights and Assumptions

Insight or Assumption	Screening Basis	Potentially Significant?	Disposition
On a Loss of Preferred Power (LOPP) all PSW pumps will need to restart and will be supplied by the non-safety related diesel generators.Because the PSW pumps are not on an uninterruptible power supply, the pumps will lose power on a LOPP. When the diesel generators start, the pumps will be demanded to restart.	Modeling preference	No	
The suggested mode of operation is for one PSW pump to be running on each train.	Modeling preference	No	
Compressor power.R12 and R13 systems are assumed to power SA compressors. This is pending system engineer confirmation.	Conservative assumption on design	No	
Cross-tie modeling.The cross-tie line and valves between the compressors are not modeled. Since one compressor train meets the success criteria. It is judged that the cross-tie line is non risk significant.	Modeling preference	No	
One manual valve of each compressor train is common cause modeled. Due to lack of specific valve data at this time, it is judged that model two of these manual valves represents reasonable risk contributions and insights at this time.	Modeling preference	No	
Due to the nitrogen supply is required to make up for operational losses, preventative maintenance is assumed to be performed during refuling outages.	Modeling preference	No	
It is assumed that during power operation, the nitrogen supply pressure from the CIS to the low and high pressure consumers is monitored in the MCR and alarmed if the pressure falls below the established setpoint.	Modeling preference	No	

Table 18-4

Analysis of Insights and Assumptions

Insight or Assumption	Screening Basis	Potentially Significant?	Disposition
The maintenance unavailability at power due to corrective maintenance for safety and non-safey related HPNSS components are judged to be negligible. As such, unavailability due to testing and maintenance on this system is currently not included.	Modeling preference	No	
The normal supply path for nitrogen from the CIS system is constantly checked upon making up for nitrogen losses in the supported equipments. As such, the availability test of this normal nitrogen supply is not needed.	Modeling preference	No	
Valves F707A, F707B are excess air flow check valves. They are only control nitrogen flow to PIS, screened out as non risk significant contribution valves.	Modeling preference	No	
Island-mode operation is not credited.In the Rev. 2 PRA model update. This assumption is conservative since the main generator can supply AC power to UATs	Modeling preference	No	
This revision only includes a simplified switchyard modeling.For Rev. 2 PRA model update, the switchyard design is not available. Therefore, the loss of power supply from the 500kV and 230kV switchyards is modeled with simple gates which act as surrogates for any potential switchyard design.	Modeling preference	No	
Transformers are assumed to be self-cooled. It is assumed that the generic transformer failure data includes the self-contained cooling system components.	Conservative assumption on design	No	

Table 18-4

Analysis of Insights and Assumptions

Insight or Assumption	Screening Basis	Potentially Significant?	Disposition
Auto-transfer control logic is assumed to use relays.Currently the auto-transfer control logic for the interlocked circuit breakers has not been designed. Due to the extra time required to initiate control signals via the NDCIS system, it is assumed that the design of the auto-transfer control logic will follow the traditional relay designs instead of using NDCIS.	Conservative assumption on design	No	
Auto-transfer from normal to alternate power supply is modeled.The interlocked circuit breakers are going to be designed with a break-before-make auto-transfer from the normal to alternate power supply. There could be a loss of power supply to the buses for several cycles.	Conservative assumption on design	No	
CCF of bus failures is not modeled since it is deemed to be negligible.	Modeling preference	No	
Cooling of switchgear and other main AC power system components except diesel generators is assumed to be not required.There are numerous main ac power system components located throughout the various buildings in the plant. Each building has different heating, ventilation, and air conditioning (HVAC) support systems.	Conservative assumption on design	No	
It is assumed that the PG buses R11-0000A1 and R11-0000A2 and PIP-A bus R11-1000A3 are powered from 125VDC switchgear control power from nonsafety-related DC bus A3. Similarly, it is assumed that the PG buses R11-0000B1 and R11-0000B2 and PIP-B bus R11-100B3 are powered from 125VDC switchgear control power from nonsafety-related DC bus B3	Conservative assumption on design	No	
The non-segregated phase bus duct or cable buses are not modeled.	Modeling preference	No	

Table 18-4

Analysis of Insights and Assumptions

Insight or Assumption	Screening Basis	Potentially Significant?	Disposition
Under-voltage and synchronization relays are not included in the component list.Due to the lack of design, relays are not included in the component list although they are included in the PRA model.	Modeling preference	No	
CCFs of bus and transformers are not modeled since it is assumed to be negligible.	Modeling preference	No	
Fuel building power centers are assumed in the PRA model.Based on the discussion with design engineer, the fuel building power centers should also be powered from the PIP buses, which are shown in the simplified drawing.	Conservative assumption on design	No	
Based on the one-line diagrams, default alignments for the isolation power buses and FMCRD power centers are assigned to either PIP bus 1000A3 or 1000B3 as normal power supply. The loss of normal power supply will initiate the auto-transfer to the alternate power supply	Modeling preference	No	
The breaker auto-transfers are assumed to be controlled by relays. Based on the communication with design engineers, the breaker auto-transfers are assumed to be controlled by relays although there is no final design yet.	Conservative assumption on design	No	
The nonsafety-related 125 VDC control power to circuit breakers is assumed to be provided by R16 system. The loss of control power will not allow auto-transfer.	Conservative assumption on design	No	
The safety-related uninterruptible 125 VAC control power to circuit breakers is assumed to be provided by R13 system. Each isolation power center breakers are controlled by its corresponding UPS division.	Conservative assumption on design	No	

Table 18-4

Analysis of Insights and Assumptions

Insight or Assumption	Screening Basis	Potentially Significant?	Disposition
Under-voltage and synchronization relays are not included in the component list. Due to the lack of design, relays are not included in the component list although they are included in the PRA model.	Conservative assumption on design	No	
It is assumed that preventive and corrective maintenance actions that can make a bus unavailable are avoided while the plant is at power.	Modeling preference	No	
The UPS system is assumed to provide power to its loads all the time. All the circuit breakers (not shown in the component list) are assumed to be closed and the only failure mode included in this model is failure to remain close (spuriously open).	Modeling preference	No	
Operators can use the manual bypass switches to connect the UPS buses to the available 480 VAC power source if the static switches fail to successfully transfer. However, this operator action is not modeled for conservatism	Modeling preference	No	
A safety-related ventilation system is not required for the batteries to perform their safety-related functions. However, battery rooms are ventilated by a system designed to remove the minor amounts of gas produced during the charging of batteries.	Modeling preference	No	
The standby battery chargers are not modeled due to the fact that they are not normally aligned. It is assumed that operator actions are required to change the alignment from normal to standby battery chargers.	Modeling preference	No	
The unavailability of buses, batteries and battery chargers due to test and maintenance is assigned a value of 5E-4.This value is taken from the EPRI URD unavailability for any major component of a standby safety system. AP1000 DC and UPS system PRA models used this value.	Data	No	

Table 18-4

Analysis of Insights and Assumptions

Insight or Assumption	Screening Basis	Potentially Significant?	Disposition
The super component, DG, is defined as the combination of the diesel engine(s) with all components	Modeling preference	No	
UV relays are assumed to be used to control the DG start function while failure of the DG instrumentation, control logic, and the attendant process detectors for system initiations, trips, and operational control are included in the DG failure to start or run events.	Conservative assumption on design	No	
No restoration error is modeled for diesel generator after test and maintenance.Diesel generators typically have rigorous post-maintenance test requirements. Therefore, it is reasonable to assume that its restoration error after T&M is negligible or is included in the generic data. Restoration errors are modelled for the fuel oil transfer system.	Modeling preference	No	
Although it is likely for the operators to manually start the DG after some auto start failures, no operator action is modeled in the DG system in this revision for simplicity, which is conservative.	Modeling preference	No	
Only one pair of load shedding failures from each DG has been modeled with CCF, which is representative. Similarly, only one pair of the 6 air-controlled dampers is modeled.	Modeling preference	No	
It is assumed that the start signal for diesel fuel oil transfer pumps is supplied by the level switch in the day tank, not from the NDCIS system.	Conservative assumption on design	No	

Table 18-4

Analysis of Insights and Assumptions

Insight or Assumption	Screening Basis	Potentially Significant?	Disposition
It is assumed that the start signal for the DG room ventilation and cooling system components is supplied by the DG start signal, not from NDCIS system. Since the normal ventilation subsystem is running, no start signal is required	Conservative assumption on design	No	
Although the relays have not been designed, it is reasonable to assume that the same UV relay used to control the auto-transfer of the DG breakers should be used to auto-start the diesel. Therefore, the UV relays modeled in MVDS (R11) are not modeled	Conservative assumption on design	No	
Currently assumed that the valves in the containment vent pathways are air-operated valves. Could potentially be nitrogen-operated, which would require a change in the dependency (loss of air to loss of N2) and perhaps the demand and transfer failure rates	Conservative assumption on design	No	
To model the isolation of a line break outside containment (BOC) in the isolation condenser system, a generic ICS loop was used. Since the probability of an ICS BOC is very low an assumption was made that only one break would occur at a time.	Data	No	
The parallel containment venting flowpath containing valves F010, F014, and F015 is not modeled because it is a test or sample line. The diameter of this line is 25mm compared to the 350mm - 500mm diameter of the primary flowpath	Modeling preference	No	
The VB isolation valves (ISVs) are assumed to be a diverse, redundant, passive, process-actuated check-valve type isolation valve for the vacuum breaker.	Conservative assumption on design	No	
It is assumed that operator fail to restore maintenance valve to "open" after outage maintenance is negligible. This is based on tech spec mandated operability checklists and main control room indication and alarm before startup is allowed.	Modeling preference	No	

Table 18-4

Analysis of Insights and Assumptions

Insight or Assumption	Screening Basis	Potentially Significant?	Disposition
The failure of the common IC/PCC pools due to catastrophic rupture is not considered in the model. The only plausible failure mechanism is due to external events, such as a beyond design basis earthquake.	Modeling preference	No	
The unavailability is assumed to be 8 hours a year for each one of the PCC loops. This number is used because tech specs limits the out of service time to 8 hours, and all 6 loops are required to be operable.	Data	No	
The FPS system has two jockey pumps (20gpm) and two booster pumps (1000 gpm). No flow from these pumps is credited for either IC-PCCS cooling, or RPV injection. The pumps are not included in the model at all.	Modeling preference	No	
No power or signal transmission dependencies are modeled for the two diesel driven FPS pumps. The system is designed so that makeup to the FAPCS can be provided by the diesel driven pumps without the need for any electrical power source.	Modeling preference	No	
In RPV makeup mode, FPS injection is assumed to fail if either FAPCS check valves F331A/B fails to close. Flow will be diverted upstream into the FAPCS system and not into the RPV.	Modeling preference	No	
FPS pumps are assumed to be out of service for maintenance one week per year.	Data	No	
The FPS system will have several normally locked open manual valves throughout the system. Though some may impact system success if mispositioned, none of the normally locked open manual valves are included in the model.	Modeling preference	No	

Table 18-4

Analysis of Insights and Assumptions

Insight or Assumption	Screening Basis	Potentially Significant?	Disposition
FPS pumps, as well as discharge and header check valves are assumed to be tested quarterly. This is likely conservative. The header check valves will open on the operation of any FPS pump (including the jockey and booster pumps).	Modeling preference	No	
The primary FPS water tanks have level indication in the control room. Makeup water level is also assumed to have indication in the control room. For determining failre rate data, the testing interval for these tanks is assumed to be 24 hours.	Data	No	
The DPV spurious opening frequency was calculated in DCD Chapter 15 as 5.75E-04/year. The check valve internal rupture (shear disk rupture) failure rate is assumed; considering a verification of the integrity of the nozzle is performed at every refueling, the annual failure rate is 4.4E-5, based on data in Appendix A of Chapter 1 of the EPRI ALWR URD (Reference 2.3-3).	Conservative assumption on data	No	
The frequency for a RWCU break outside containment is assumed to be the same as that for a feedwater line break and is 3.4E-03yr.	Data	No	
The MGL method is considered the most appropriate because of its combination of conservatism and its ability to explicitly model contributions of subgroup failures.	Modeling preference	No	
The use of generic data for most of the electrical and mechanical components is justified by the following: In this phase, the specific types of components that will be purchased are not defined. These generic data are representative of components used in previous BWR plants.	Modeling preference	No	

Table 18-4

Analysis of Insights and Assumptions

Insight or Assumption	Screening Basis	Potentially Significant?	Disposition
Reactor Water Level Instrumentation Failures A possible local effect could be generated by a LOCA if the break is close to the instrumentation piping. For this case, only one division is expected to be affected, so no significant contribution to core damage frequency is expected.	Modeling preference	No	
LOCAs Inside Containment A very small LOCA (G1) is defined as a pipe break or component failure that results in a loss of primary coolant between 10 to 100 gpm. It is assumed that the condensate /feedwater makeup capacity is at least 100 gpm. A reactor trip would not be required for a very small LOCA.	Modeling preference	No	
Accident Sequence Methodology If there is no core damage until more than 72 hours from the initiating event, there is sufficient time to implement recovery actions, including repair of failed equipment.	Modeling preference	No	
Success Criteria Identification Typically, credit is not taken for the benefits of partial functioning of safety functions in the PRA.	Modeling preference	No	
No credit is taken for manual RPS actuation by the operators during an initiating event.	Conservative assumption on design	No	
Feedwater Pump Run Back Assumptions: This action is automatic, and no manual recovery is credited in the ATWS sequences. Conservatively, the failure of this function is assumed to lead to core damage due to a failure to initially reduce reactor power during an ATWS condition.	Conservative assumption on design	No	

Table 18-4

Analysis of Insights and Assumptions

Insight or Assumption	Screening Basis	Potentially Significant?	Disposition
Standby Liquid Control System Assumptions: Conservatively, no back up or long term recovery of this function is considered. The failure of this function is assumed to lead to core damage due to failure to successfully control power.	Conservative assumption on design	No	
RPV Overpressure Protection Assumptions: Conservatively, an overpressurization of the RPV is assumed to result in a reactor vessel rupture (RVR) and a transfer to the RVR event tree. No credit is taken for operator action to open an SRV.	Conservative assumption on design	No	
RPV Overpressure Protection During ATWS Assumptions: Conservatively, the failure of this function is assumed to lead to core damage due to RPV rupture and re-criticality at low RPV pressure.	Conservative assumption on design	No	
Isolation Condensers Assumption: Although MAAP analysis shows that only 2 ICs are needed to control reactor pressure after one hour in a transient initiating event, the PRA model does not take credit for partial functioning or reduced success criteria that are based on timing.	Conservative assumption on design	No	
FW Injection Assumptions: It is conservatively assumed that if top event QT is failed, then top event UF is unavailable. Operator actions for monitoring and controlling feedwater and condensate pumps are within the normal process for responding to a feedwater transient.	Conservative assumption on design	No	
ADS Assumptions: No credit for pressure reduction is taken for SRVs opening if the DPVs have failed.	Modeling preference	No	

Table 18-4

Analysis of Insights and Assumptions

Insight or Assumption	Screening Basis	Potentially Significant?	Disposition
Containment Venting Assumptions: A low pressure injection source must be available after containment venting to ensure RPV water level is maintained. This dependency is captured within the event trees.	Modeling preference	No	
In the case of failure of at least one SRV to open and relieve high RPV pressure, the situation is assumed to be a Reactor Vessel Rupture (RVR), and it is transferred to the RVR event tree.	Modeling preference	No	
In some sequences, there is a loss of all high pressure injection, and the vacuum breakers fail to open after ADS. This leads to containment failure due to excessive pressure differential between the drywell and the wetwell airspace. This is assumed to lead to core damage.	Modeling preference	No	
After a successful scram and a loss of feedwater, the power conversion system is assumed to be failed due to the loss of feedwater.	Conservative assumption on design	No	
Sequences involving a LOCA and a failure to scram are assumed to lead directly to core damage.	Conservative assumption on design	No	
FDW is assumed failed for a large break in either FDW line due to loss of FDW suction inventory.	Conservative assumption on design	No	

Table 18-4

Analysis of Insights and Assumptions

Insight or Assumption	Screening Basis	Potentially Significant?	Disposition
The flow rate at reactor pressure for a medium break liquid LOCA is greater than the CRD makeup capacity. Depressurization is needed for GDCS injection to prevent core uncovery. It is assumed that FAPCS in the LPCI injection mode, which takes suction from the suppression pool, is lost on suppression pool low water level, before the level of water outside the vessel can maintain the core covered. Therefore, FAPCS in the suppression pool cooling mode is not available in those cases. Due to water level in the vessel being below Level 3, the upper suction line to RWCU/SDC is not available. Therefore, it is assumed that shutdown cooling is not available.	Modeling preference	No	
The combination of an ATWS with an unisolated break outside containment is assumed to lead to core damage.	Modeling preference	No	
A high DW temperature could be caused by accidents such as 1) LOCAs, 2) inadvertent opening of one DPV, or 3) loss of the drywell cooling system. However, the instrumentation is assumed to be designed for the maximum temperature attainable in the DW, so no instrumentation effects should occur.	Modeling preference	No	
A line break in FDW Line A could affect the functioning of RWCU and FAPCS lines that connect to the line. Therefore, it is assumed that RWCU shutdown cooling, FAPCS LPCI, and FPS injection are not available. In addition, containment venting is assumed to be ineffective because low pressure injection from LPCI or FPS are unavailable to maintain RPV water level. Short-term core damage sequences are considered to be Class V, (containment bypass) sequences.	Modeling preference	No	

Table 18-4

Analysis of Insights and Assumptions

Insight or Assumption	Screening Basis	Potentially Significant?	Disposition
ADS inhibit also prevents uncontrolled reactor depressurization and the subsequent GDCS injection which could lead to boron washing out of the RPV. It is assumed that without ADS inhibit the resulting boron dilution causes recriticality.	Modeling preference	No	
With successful SLCS injection, core cooling can be performed: (1) by ICS if all the valves that are open also close correctly, (2) by the FDW or CRD with the heat removal by the FAPCS in the cooling mode; or, (3) by RWCU/SDC if isolation signals are inhibited and the filters are bypassed. In case of failure of high pressure systems, core damage is assumed because the reactor is depressurized, and reactivity control is not credited in preventing core damage because of the potential for boron dilution.	Modeling preference	No	
ATWS sequences with failure of the Feedwater runback function are assumed to lead directly to core damage.	Modeling preference	No	
Sequences involving failure of SLCS are assumed to lead directly to core damage.	Modeling preference	No	
Because combustible gas deflagration cannot be excluded when the containment atmosphere is deinerted, this analysis conservatively assumes that all core damage events during this deinerted window will lead to containment failure.	Modeling preference	No	
The TSL release category provides a source term that exceeds that associated with normal operation because of the severe accident conditions within the containment. It is assumed that the leakage area corresponds to the Technical Specification allowable containment leakage rate of 0.5% of containment air volume per day at rated pressure.	Modeling preference	No	

Table 18-4

Analysis of Insights and Assumptions

Insight or Assumption	Screening Basis	Potentially Significant?	Disposition
The leakage path was conservatively assumed to occur between the drywell atmosphere and environment. Thus, no credit is taken for source term reduction if the leakage could be affected by potential refueling pool scrubbing or the reactor building HVAC system.	Modeling preference	No	
Air-operated vent valve opening is modeled using the same functional top gate as in the Level 1 analysis: WV-TOPVENT. Re-close of the vent valves is not modeled; the release is assumed to continue once initiated.	Modeling preference	No	
The severe accident progression is assumed to lead to core relocation onto the drywell floor. After RPV failure, the corium cannot be credibly postulated to form a critical, ex-vessel configuration. Therefore, reactivity control methods are not required for severe accident mitigation and no reactivity control equipment is considered in the severe accident survivability evaluation.	Modeling preference	No	
The release category “Bypass” represents those sequences in which containment isolation has not occurred due to failure of the Containment Isolation System (CIS) function. Thus, there is a direct path from the containment atmosphere to the environment when the severe accident is initiated. To determine the source term, a large diameter pipe opening was assumed from the time of accident initiation.	Modeling preference	No	
Many evacuation related characteristics (local roads, population demographics, emergency services) are site specific. The evacuation parameters used in this study are conservative assumptions in that no evacuation or relocation assumed and no sheltering is assumed. The public is assumed to continue normal activity during the reactor accident in this bounding analysis.	Modeling preference	No	

Table 18-4

Analysis of Insights and Assumptions

Insight or Assumption	Screening Basis	Potentially Significant?	Disposition
For each source term, the release is modeled to occur at ground level. The thermal content of the plume is assumed to be the same as ambient. These assumptions are conservative for early fatalities	Modeling preference	No	
In Table 12.3-1, another two new fire areas are created: FFPE and FSWYD. The first one assumes the fire pumphouse enclosure for fire protection system. Fire area FSWYD is used to evaluate a postulated fire in the switchyard area that is conservatively assumed to result in a loss of preferred power.	Modeling preference	No	
Currently there is no detailed circuit analysis. However, based on the DCIS design basis, the MCR controls will be connected to the back panel rooms (unaffected by a main control room fire) via fiber cables and that the loss (including melting) of the cables or visual display units (VDUs) will not cause inadvertent actuations nor affect the automatic actions associated with safety and non-safety equipment.	Conservative assumption on design	No	
ESBWR plant has a passive design that the safety systems do not have active components such as the high-pressure injection pumps in the traditional plant designs. For the high/low pressure interfaces, multiple check valves are included which prevent the opening of the path even if a spurious actuation should occur after a fire. DCD Tier 2 section 7.6.1 describes the HP/LP system interlock function.	Modeling preference	No	
A fire in the Service Water / Water Treatment Building (fire area F7300) is assumed to result in loss of service water. A fire propagation case with both F3301 and F3302 would result in a loss of service water. However, it is determined that the loss of RWCU initiator would be bounding.	Modeling preference	No	

Table 18-4

Analysis of Insights and Assumptions

Insight or Assumption	Screening Basis	Potentially Significant?	Disposition
The plant general arrangement drawings with component locations are used to perform a reasonable cable routing list. A list of cables is generated from the support table in system model database that includes the all modeled supports for PRA components included in the system models. This list should have captured the majority of cables, especially the risk-important components.	Conservative assumption on design	No	
Fibers are assumed to connect RMUs. Hard wires are assumed to connect the components to RMUs. To prevent spurious actuations, load drivers are assumed to be located in different fire areas.	Conservative assumption on design	No	
For QDCIS cables, it will typically originate from the QDCIS divisional room in the control building and pass through its own divisional duct bank, then connect to its divisional cable chase in the reactor building.	Modeling preference	No	
For NDCIS cables, it will typically originate from the NDCIS rooms in the control building and pass through the nonsafety-related divisional tunnel and connect to rooms in the reactor building, turbine building, or electrical building.	Modeling preference	No	
If the NDCIS cable has to pass through the divisional rooms in reactor building, it is assumed that QDICS Div 1 and 3 rooms can be used for NDCIS Div A and Div 2 and 4 used for Div B.	Conservative assumption on design	No	
Within each plant, the likelihood of fire ignition is the same across an equipment type. For example, pumps are assumed to have the same fire ignition frequency regardless of size, usage level, working environment, etc.	Data	No	

Table 18-4

Analysis of Insights and Assumptions

Insight or Assumption	Screening Basis	Potentially Significant?	Disposition
Although the ESBWR plant may be located with the existing nuclear power plants, the plant location weighting factors are assumed to be 1 for simplicity so that there is no need to count the components in those existing plants.	Modeling preference	No	
Since no history on the maintenance activity is available, the weighting factor evaluation is simplified. It is assumed that all compartments have the same transient fire influencing factors. This is conservative since the high risk areas are going to have tighter controls. Potential exceptions are the main control room (fire area F3270) and the turbine building general area (fire area F4100). The main control room typically has high occupancy and the bin 7 (transient) count is increased to 10. The turbine building general area covers a large portion of the turbine building and would expect high maintenance activity and high occupancy all the time. For conservatism, the weighting factor for this area (for bins 36 and 37) is increased by a factor of 10.	Conservative assumption on design	No	
The bins for cable fires, cable run and junction boxes are estimated with the cable routing information generated in the cable selection task.	Conservative assumption on design	No	
For main control room and DCIS room fire ignition frequency calculations, additional non-PRA transformers, cabinets, and AHUs are counted, which are shown on the general arrangement drawings. This is conservative since the total counts of those components are based on the PRA components only.	Modeling preference	No	

Table 18-4

Analysis of Insights and Assumptions

Insight or Assumption	Screening Basis	Potentially Significant?	Disposition
It is further assumed that the pipe routed to or from the equipment would follow certain logical paths. For example, pipe routing is through pipe chases in battery rooms instead of routing the pipe through the battery room. Another logical path would be to take the shortest path which reduces piping and fabrication cost.	Modeling preference	No	
Water in a stairwell or propagating into a stairwell preferentially continues to travel down the stairwell as opposed to propagating under a door leading outside the stairwell.	Modeling preference	No	
Concurrent flooding events from different sources are not considered in the flooding analysis.	Modeling preference	No	
Components that are environmentally qualified inside containment are considered to be invulnerable to the effects of flooding.	Modeling preference	No	
It is assumed that insulation properties are not lost at any point of cable routing and that interaction with water can only occur at termination points that are not environmentally qualified.	Modeling preference	No	
Fire doors are not necessarily watertight.	Modeling preference	No	
Walls are assumed to be capable of withstanding the expected maximum flood loading. Therefore, walls are assumed to remain intact throughout a flooding event.	Modeling preference	No	

Table 18-4

Analysis of Insights and Assumptions

Insight or Assumption	Screening Basis	Potentially Significant?	Disposition
It is assumed that electrical circuit fault protection has been designed to provide protection for plant electric circuits via protective relaying, circuit breakers, and fuses. Therefore, loss of a component due to flooding will not result in the loss of the bus that supplies power to the affected component.	Modeling preference	No	
Safety-related electrical equipment will be qualified for environmental conditions in which they are required to function. Containment isolation valves located inside containment are not evaluated for flood-induced failure with respect to containment isolation because they will be qualified for the environmental conditions in which they are required to function.	Modeling preference	No	
For floor drains, appropriate precautions such as check valves, back flow preventers, and siphon breaks are assumed to prevent back flow and any potential flooding.	Modeling preference	No	
Dry pipe systems (such as a pre-action FPS system) are not modeled as flood sources due to the low frequency of a failure of the dry pipe coincident with spurious opening of the actuation valve.	Modeling preference	No	
Flooding in the containment during shutdown will not affect ICS components or the DPVs because these components are relatively high in the containment.	Modeling preference	No	
EF2 and EF3 tornados would exceed the wind speed design of non-seismic structures but would not affect RTNSS or Seismic Category I or Cat II structures. Therefore, for EF2 and EF3 tornados the equipment located in non-seismic structures or in the yard will be assumed to fail.	Modeling preference	No	

Table 18-4

Analysis of Insights and Assumptions

Insight or Assumption	Screening Basis	Potentially Significant?	Disposition
The model conservatively assumes that both SLCS trains are required for success because of uncertainties associated with the SLCS flow model. The result of this conservatism is that flooding in the reactor building at +17500 elevation or failure of a SLCS pipe and failure of rods to insert results in core damage.	Modeling preference	No	
Spent fuel cooling: This function will be maintained during shutdown modes just as it will during full power modes. It is assumed to have no significant impact to the shutdown model	Modeling preference	No	
Reactivity control is assumed to have no significant impact on the shutdown PRA model.	Modeling preference	No	
Power supplies are not modelled in RPS since de-energizing circuits leads to success.	Modeling preference	No	
Power availability: Modeled as it is in the full power model. 'Loss of Preferred Power' is a shutdown initiating event due to it leading to a loss of decay heat removal.	Modeling preference	No	
Containment: Containment is open for much of the shutdown model, and it not credited in the model for the time it is maintained. All core damage sequences modeled in the shutdown PRA are assumed to lead to a direct containment bypass.	Modeling preference	No	
If the reactor well is flooded (Mode 6-Flooded), the risk associated to loss of decay heat removal has been judged to be negligible because in addition to RWCU/SDCS, FAPCS can be aligned to cool the reactor well water, constituting a valid alternative for RWCU/SDCS, thus reducing the probability of losing the decay heat removal function.	Modeling preference	No	

Table 18-4

Analysis of Insights and Assumptions

Insight or Assumption	Screening Basis	Potentially Significant?	Disposition
The large amount of water stored above the core assures core cooling during a long period of time. This time would be significantly longer than 24 hours. This long period could be used to establish an adequate path from an external water source to the reactor well. CRD pumps, FAPCS pumps, condensate pumps, or firewater pumps could provide this makeup function. The long period of time available makes it practically certain that sufficient inventory can be supplied. Therefore, the loss of decay heat removal is not analyzed in detail for the case when the reactor well is flooded (Mode 6-Flooded).	Modeling preference	No	
For the other shutdown modes (Modes 5 and 6 with the reactor well unflooded), it is assumed that one RWCU/SDCS train is sufficient to remove decay heat to prevent reactor coolant boiling.	Modeling preference	No	
It is assumed that both RWCU/SDCS trains are running, because the time periods in which only one is running occurs when the reactor well is flooded. Consequently, failure of one of the trains is not considered an initiating event.	Modeling preference	No	
If the reactor well is flooded, the water inventory stored above the core is assumed to be sufficient to flood the drywell and the vessel well above the TAF if the two lower drywell access hatches are closed at the time of the event or they are manually closed before the water level in the drywell reaches the elevation of the hatches.	Modeling preference	No	
The same LOCA scenarios modeled for Mode 6 are modeled for Mode 5 with and without an intact containment. Steam Line LOCAs and DPV line LOCAs are assumed to pose negligible risk. The LOCA frequencies for these are lower than GDCS and feedwater, and the resulting LOCAs don't disable any makeup function.	Modeling preference	No	

Table 18-4

Analysis of Insights and Assumptions

Insight or Assumption	Screening Basis	Potentially Significant?	Disposition
An evaluation of system pipe drawings showed two potential draindown paths due to misalignment. Both lines have 20mm diameters and are assumed to be too small to be considered and initiating event	Modeling preference	No	
Draining the RPV during FMCRD maintenance has been evaluated, but is not considered a shutdown PRA initiating event. With the tools used for the actions, and the controls in place to monitor the evolution, the chance of a significant RPV leak due to the activity is assumed to be negligible.	Modeling preference	No	
Once the event has been detected, the plant operator must correctly diagnose the situation, make the decision to close the hatches, gain access to elevation –6400 mm in the reactor building, and manually close the equipment hatch and the personnel air lock. It is assumed that during the outage, personnel will be continuously located in the area of the doors.	Modeling preference	No	
It is assumed that during the entire shutdown period a single CRD pump is running (providing purge flow to FMCRD and/or the RWCU/SDC pumps) and the second pump is in standby, which is the same initial configuration as at full power. Additionally, it is assumed that no automatic initiation of CRD injection is available and alignment in RPV injection mode requires operator action.	Modeling preference	No	
The PRA model conservatively assumes that a single failure on either train of SLC caused core damage if the control rods fail to insert (ATWS). The CDF would be reduced by approximately 13% if either train of SLC is able to mitigate ATWS scenarios.	Results	Yes	Insight

Table 18-4

Analysis of Insights and Assumptions

Insight or Assumption	Screening Basis	Potentially Significant?	Disposition
The results indicate that the pre-initiators have a significant impact on the RAW value. This is primarily due to the large number of potential latent failures, and relatively high reliability for each of these operator actions. The post-initiator HRA screening values are relatively high. As expected, the post-initiator HRA values have a high FV, but a relatively lower impact on RAW.	Results	Yes	Insight
An increase of one order of magnitude for vacuum breaker and back-up valve failure rates, only causes the CDF to increases by approximately 10%. However, the Level 2, non-TSL frequency increases by approximately 120% over the baseline results. Steam suppression failures generally lead to core damage states against which release can not be mitigated (class ii-a and class v).	Results	Yes	Insight
Changes to squib valve failure rate, have a significant impact to CDF. As expected, increases in the failure rate of the squib valves used for the ADS cause a significant increase in the accident Class III contribution (core damage at RPV high pressure). Similarly, increases in the failure rate of the squib valves used for the GDCS cause a significant increase in the accident Class I contribution (core damage at RPV low pressure). However, an increase only to the SLC squib valves does not have a very pronounced impact on CDF.	Results	Yes	Insight
The ESBWR Level 1 PRA core damage frequency is significantly impacted if the non-safety related systems are not credited. If credit is taken for all the RTNSS systems, the focused Level 1 PRA results are reduced by almost two orders of magnitude. However, the impact to CDF can be minimized, by about one order of magnitude if one only credits the availability of the Diverse Protection System (including surrogate logic for DPS signal for MSIV isolation).	Results	Yes	Insight

Table 18-4

Analysis of Insights and Assumptions

Insight or Assumption	**Screening Basis**	**Potentially Significant?**	**Disposition**
Including the DPS and ARI functions (ARI function is used as surrogate logic for DPS MSIV isolation that is currently not modeled) with the safety-related systems allows the nTSL release frequency to satisfy the NRC goal of 1E-6/yr for LRF in the internal events, fire, and flooding Level 2 PRA models.	Results	Yes	Insight
Improving the reliability of the nitrogen accumulators by providing alarm/indication of accumulator pressure would enhance operator response to low pressure to mitigate the risk of accumulator failures.	Results	Yes	Operational Program
Improving the reliability of re-filling the pools to maintain long term ICS and PCCS availability would reduce risk of long-term sequences. Actions such as independent verification of valve position would reduce the failure probability.	Results	Yes	Operational Program

F-V and RAW Importance Measures Report

(Deleted)

Table 18-5

Internal Flooding Full-Power Importance Measure Report (Deleted)

F-V and RAW Importance Measures Report

(Deleted)

Table 18-6

Internal Flooding Shutdown Importance Measure Report (Deleted)

F-V and RAW Importance Measures Report

(Deleted)

Table 18-7

Tornado Full-Power Importance Measure Report (Deleted)

F-V and RAW Importance Measures Report (Deleted)

Tornado Full Power

Core Damage Frequency = 4.77E-11 (Deleted)

Table 18-8

Tornado Shutdown Importance Measure Report (Deleted)

F-V and RAW Importance Measures Report

Tornado Shutdown

(Deleted)

Table 18-9

Shutdown PRA Importance Measure Report (Deleted)

F-V and RAW Importance Measures Report

(Deleted)