18 PRA INSIGHTS AFFECTING ESBWR DESIGN

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18 PRA INSIGHTS AFFECTING ESBWR DESIGN

18.1 INTRODUCTION

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.

This section describes a systematic approach that is used to define insights and assumptions, identify and evaluate their significance, and to ensure that they are understood and preserved throughout the design phase of the ESBWR.

18.2 PRA ASSUMPTIONS

18.2.1 Understanding the Effects of Uncertainties on the PRA Model

Uncertainty exists in the PRA model where there is no consensus approach or modeling method, and where the choice of approach or modeling method could have an effect on the risk profile such that it influences a risk-based decision made using the PRA. In response to a source of uncertainty, an assumption is created to acknowledge that a best-estimate approach is used, but a different reasonable alternative assumption could produce different results.

During the development phase of the ESBWR, aspects of the design that are not fully complete are treated in the PRA model by using modeling assumptions. As such, it is necessary to identify these assumptions and assure that a process exists to maintain their validity. Different PRA 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. Therefore, it is important to identify significant, or "key," assumptions and determine if the use of alternative methods or numbers could affect the results and insights of the PRA.

Uncertainties can be categorized into distinct groupings for identification and treatment. Parametric 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. The effects of uncertainties are assessed by performing quantified analyses or qualitative arguments with conclusions that provide reasonable assurance that, 1) the PRA results represent the as-to-be built plant, and 2) key assumptions are preserved throughout the design phase to ensure they are assessed when making design changes.

18.2.2 Identifying Key Assumptions

An assumption is termed "key" if other reasonable alternatives may exist whose results are significant enough to affect the PRA results and conclusions. A systematic method is used to identify the key assumptions of the PRA model. Sections 1 through 16, and 21 of NEDO-33201 are reviewed to identify instances where parametric, modeling and completeness uncertainties require the use of assumptions. This provides a comprehensive accounting for assumptions from each phase of the PRA development, including the severe accident analysis. Assumptions are identified in each step of the PRA development process. This includes initiating events, accident sequences, success criteria, failure probabilities, human error probabilities, and quantification methods. In addition, each system fault tree logic gate is assessed to identify assumptions that are used, to justify items that are included as well as those that are excluded from the fault tree models. After these assumptions are identified, they are evaluated by a qualitative assessment or sensitivity study to determine their relative significance in terms of the NRC Safety Guidelines for CDF (<1% change of 1E-4), LRF (<1% change of 1E-6), and CCFP (<0.01 increase). The final result is documentation of all assumptions and a list of key assumptions (i.e., risk significant) and their dispositions, which are as follows:

- Design Requirement: an assumption that requires specific design details be preserved to maintain its validity. Design Requirement key assumptions are controlled by the design organization (GEH).
- Operational Program: an assumption that requires specific operational procedures or training be served to maintain its validity. These key assumptions are implemented in the normal operations phase by the licensees.

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-33201 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. 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. 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.

18.4 MAINTAINING RISK INSIGHTS AND ASSUMPTIONS

As discussed above, a systematic process is necessary to ensure the continued validity of the model, the following describes the methodology employed.

(1) Review the NEDO-33201 Sections 1 through 16 and 21 to identify assumptions and insights.

  a. Review the section to identify risk insights and assumptions as follows:

    i. Areas where certain design features are the most effective in reducing risk with respect to operating reactor designs;

    ii. Major contributors to risk, such as hardware failures and human errors;

    iii. Major contributors to maintaining the "built-in" plant safety and ensuring that the risk does not increase unacceptably;

    iv. Major contributors to the uncertainty associated with the risk estimates; and

    v. 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.

  b. Document the identified insights and assumptions.

(2) Review the system fault tree models, identify and document assumptions for each gate.

  a. For model updates, review revisions to fault tree gates to identify new assumptions.

(3) Using the flow chart in Figure 18-1, evaluate key insights and assumptions.

  a. Determine if evaluation is to be qualitative or quantitative;

  b. If qualitative evaluation is sufficient, screen against criteria

  c. If quantitative evaluation required or desired, evaluate against the following criteria:

    i. CDF/LRF importance measures

    ii. Sensitivity study cases

  d. Document results appropriately.

(4) Document insights and assumptions identified in subject NEDO-33201 Sections 1 through 16 and Section 21 as follows:

  a. Summarize all insights and assumptions captured using this process.

  b. Add to or add a new subsection for "Key Insights" in the subject NEDO-33201 section to capture all key insights and assumptions

(5) Review insights and assumptions with respect to CDF, LRF, and CCFP.

  a. Items that are important from a system standpoint, but do not significantly affect CDF, LRF, or CCFP, are retained in system information but are not "key" insights with respect to the DCD application.

b. Summarize results in this NEDO-33201 Section 18, Table 18-1.

(6) Update DCD Tier 2 Chapter 19, Table 19.2-3, with the key insights and assumptions

Table 18-1

Risk Insights and Assumptions

	Insight or Assumption	NEDO-33201 Section	Disposition
1	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 nominal allowed leakage. The ESBWR is designed to minimize the effects of pressurization due to direct containment heating (suppression pool – DCD Tier 2, Subsection 6.2.1.1), ex-vessel steam explosions (GDCS pool spillover pipes– DCD Tier 2, Subsection 6.2.1.1.10.2), and core-concrete interaction (BiMAC – DCD Tier 2, Subsection 19.3.2.6). The ESBWR containment is designed to a higher ultimate pressure than previous BWR containment designs.	21.1	Insight
2	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.	10.4	Insight
3	The ESBWR incorporates redundancy and diversity in its design principles, and has used PRA insights during development to identify potential risks and to address them in the design phase. As such, the risks of core damage and offsite radiological consequences are very low. In addition, the risk profile is balanced such that there are no individual component failures or operator errors that contribute a proportionally significant risk. The relative risk significance of individual risk contributions from ESBWR SSCs and operator actions are very low, and are on the same order of magnitude, in some cases, of events that were previously excluded in LWR PRAs.	17.1	Insight
4	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.	2.2	Insight

Table 18-1

Risk Insights and Assumptions

	Insight or Assumption	NEDO-33201 Section	Disposition
5	Sensitivity study results indicate that changes in test and maintenance unavailability do not significantly impact the CDF or insights.	11.3	Insight
6	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 NEDO-33201 Section 2, Table 2.3-2. Sensitivity study results indicate that changes in the LOCA frequencies have the potential to impact CDF, but still maintain significant margin below the NRC safety goal guidelines.	2.2	Operational Program – Site Baseline PRA
7	Sensitivity study results indicate that changes in the human error failure probabilities, particularly pre-initiators, have the potential to impact CDF, but still maintain significant margin below the NRC safety goal guidelines.	11.3	Operational Program – Human Factors Engineering
8	Sensitivity study results indicate that squib valve failure rate estimates have the potential to impact CDF, but still maintain significant margin below the NRC safety goal guidelines.	11.3	Operational Program – Maintenance Rule
9	If automatic isolation fails to isolate an RWCU/SDC line break outside of containment, manual action to isolate the line is modeled in sequences that allow successful low pressure injection.	3.3	Operational Program – Procedure Development
10	If the containment is breached or is vented, GDCS injection fails to be sustained in long-term sequences because condensate make-up to GDCS from PCCS is not available. An external injection source must be available in these sequences to ensure adequate core cooling.	3.3	Operational Program – Procedure Development

Table 18-1

Risk Insights and Assumptions

	Insight or Assumption	NEDO-33201 Section	Disposition
11	The design certification PRA uses conservative assumptions in the treatment of ATWS conditions. The following failures are assumed to lead directly to core damage: • Feedwater runback • ADS Inhibit • Level control using Feedwater or CRD injection • SLC The Site Baseline PRA will be refined to reflect operating procedures that will be developed to address the response to ATWS conditions in more detail.	3.3	Operational Program – Site Baseline PRA
12	Q-DCIS and N-DCIS are designed to high standards for reliability, including very reliable hardware and high quality software. The most dominant failure modes reside in the uncertainty in the treatment of software faults, including common cause software failures that either cause demanded actions to fail, or cause spurious actions.	4.5	Operational Program – Site Baseline PRA
13	GDCS faults are dominated by common cause failures of the check valves or the squib valves in the injection and equalize lines.	4.6	Operational Program – Maintenance Rule
14	CRD injection is assumed to be functional following a containment overpressurization failure due to the separation between the dominant containment failure locations (DCD Tier 2, Appendix 19C) and the location of CRD pumps and lines. This is an important assumption, based on the containment failure analysis, which supports the use of CRD in these sequences.	3.3	Operational Program – Site Baseline PRA

Table 18-1

Risk Insights and Assumptions

	Insight or Assumption	NEDO-33201 Section	Disposition
15	The following operator actions have the highest risk importance: • Fail to recognize the need for IC/PCCS pool makeup • Fail to recognize the need for makeup after depressurization • Fail to close Lower Drywell Hatches after a LOCA during Shutdown. These operator actions are based on conservative modeling methods and none are considered to be dominant contributors to CDF or LRF.	17.1	Operational Program – Human Factors Engineering
16	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.	3.3	Design Requirement (DCD Tier 2 Subsection 9.1.3.2)
17	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–related 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.	12.10	Design Requirement (DCD Tier 2 Section 1.0, F1.2-4)
18	The exposure of the distributed control and information system (Q-DCIS and N-DCIS) equipment to heat and smoke caused by a fire in a single fire area does not cause spurious actuations that could adversely affect safe shutdown.	12.10	Design Requirement (DCD Tier 2 Subsection 9.5.1.10)

Table 18-1

Risk Insights and Assumptions

	Insight or Assumption	NEDO-33201 Section	Disposition
19	The communication links between the main control room (MCR) and the Q-DCIS and N-DCIS rooms do not include any copper or other wire conductors that could potentially cause fire-induced spurious actuations that could adversely affect safe shutdown.	12.10	Design Requirement (DCD Tier 2 Subsection 9.5.1.12)
20	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.	13.8	Design Requirement (DCD Tier 2 Subsection 3.4.1.4.3)
21	The Drywell Floor Drain Sump channels, which allow leakage on the lower drywell floor to flow into the sump, will prevent any molten debris, which reaches the inlet, from entering the sump.	21.5	Design Requirement (DCD Tier 2 Subsection 6.2.1.1.10.2)
22	Closure of both the equipment hatch and the personnel hatch can be performed from outside the lower drywell/containment.	16.3	Design Requirement (DCD Tier 2 Section 1.0, F1.2-2)
23	DELETED		
24	The IC/PCCS Pool valves that provide make-up water from the equipment storage pool have DPS controls and are powered from a reliable source of power, which is capable of long-term support.	4.2	Design Requirement (DCD Tier 2 Table 7.8-3 and Subsection 5.4.6.2.2)

Table 18-1

Risk Insights and Assumptions

	Insight or Assumption	NEDO-33201 Section	Disposition
25	Control logic cabinets for each of the containment vacuum breaker isolation valves must be located in separate fire zones.	4.18	Design Requirement (DCD Tier 2 Subsection 6.2.1.1.2)
26	Because of the high consequence of a RWCU/SDC line break outside containment this system is designed with an additional diverse, nonsafety-related valve that is used for line isolation. This valve is controlled by the nonsafety-related DCIS system and closes on the same signals that provide the safety-related isolation.	4.8	Design Requirement (DCD Tier 2 Subsection 5.4.8.1.2)
27	Power operated equipment and valves on lines attached to the RPV that require maintenance have maintenance valves installed such that freeze seals will not be required.	16.3	Design Requirement (DCD Tier 2 Subsection 5.2.3.1.1)
28	Separate common cause failure groups are assumed in the PRA model for safety-related versus nonsafety-related batteries and inverters.	4.15, 4.17	Design Requirement (DCD Tier 2 Subsection 7.1.1)
29	A pneumatic accumulator and check valve are required to support the remote-manual and ADS-activated functions of the valve. The accumulator and check valve ensures that the valve opens via the pneumatic operator following a failure of the pneumatic pressure source.	4.1	Design Requirement (DCD Tier 2 Subsection 5.2.2.2.2)
30	The composition of the layer of sacrificial material on the lower drywell floor that covers the BiMAC piping is designed, for the more likely severe accident sequences, to prevent melt impingement due to corium ablation, and also to prevent noncondensable gas generation in quantities that would lead to exceeding the containment ultimate pressure.	21.5	Design Requirement (DCD Tier 2 Subsection 19.3.2.6)

Table 18-1

Risk Insights and Assumptions

	Insight or Assumption	NEDO-33201 Section	Disposition
31	The 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 NEDO-33201 Appendix 8A that the frequency of ATWS sequences experiencing RPV rupture at high pressure is negligible, so only failures at low pressure were analyzed.	8A	Operational Program – Procedure Development
32	Venting is assumed to occur when the containment pressure reaches 90% of the ultimate containment strength.	3.3	Operational Program – Procedure Development
33	During shutdown conditions, a continuous fire watch is required for the following scenarios with breached fire barriers for maintenance activities: • The breaching of the fire door between fire areas F1152 and F1162 (the reactor building fire areas that house RWCU pumps). • The simultaneous breaching of multiple fire barriers that can open fire areas F3301 and F3302 (the N-DCIS room fire areas) to fire area F3100 (the corridor fire area) at the same time. • The simultaneous breaching of multiple fire barriers that can open fire areas F5350 and F5360 (the PIP electric equipment room fire areas) to fire area F5100 (the corridor fire area) at the same time. Shutdown fire risks related to the fire barriers are evaluated and managed in accordance with the outage risk management program of 10 CFR 50.65(a)(4).	12.10	Operational Program – Maintenance Rule

Table 18-1

Risk Insights and Assumptions

	Insight or Assumption	NEDO-33201 Section	Disposition
34	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).	16.3	Operational Program – Procedure Development
35	An important recovery action during shutdown is to recover at least one train after loss of both operating RWCU/SDCS trains. This is the primary method of residual heat removal. In the limiting case of loss of cooling, there are approximately 4 hours before boiling would occur. Therefore, there is ample time to restore RWCU/SDC or its supporting systems, such as Service Water or Reactor Component Cooling Water.	16.3	Operational Program – Procedure Development
36	During Mode 5, while preparing to remove the RPV head, RPV water level is raised to provide additional shielding for the personnel removing the head bolts. In BWRs, level is raised to approximately the level of the flange to maximize shielding. In the ESBWR, with its additional RPV height to accommodate the chimney, water level could be raised to a point below the vessel flange to achieve equivalent shielding protection for the workers. In addition, if water level is raised to below the ICS inlet lines, ICS can still be used to remove decay heat, in the event that shutdown cooling is lost during this time period. The duration of this configuration is estimated to be small, around 12 hours, so the overall risk contribution is small.	16.3	Operational Program – Procedure Development
37	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.	14.7	Operational Program – Procedure Development

Table 18-1

Risk Insights and Assumptions

	Insight or Assumption	NEDO-33201 Section	Disposition
38	A 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.	16.7	Operational Program – Maintenance Rule
39	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.	13.8	Operational Program – Human Factors Engineering (alarm), Procedure Development (response)
40	It is assumed that, during shutdown, manual and automatic depressurization (ADS) of the vessel are available while the vessel head is in place.	16.3	Operational Program - Technical Specification LCO 3.5.3
41	It is assumed that the actuation of the GDCS due to an RPV Level 1 water level signal is available during Mode 5 and Mode 6 Unflooded.	16.3	Operational Program - Technical Specification LCO 3.5.3
42	Procedures have provisions to prohibit coincident removal of the control rod and CRD of the same assembly.	16.3	Operational Program – Procedure Development
43	Contingency procedures provide core and spent fuel cooling mitigative actions during FMCRD replacement with fuel in the vessel.	16.3	Operational Program – Procedure Development

Table 18-1

Risk Insights and Assumptions

	Insight or Assumption	NEDO-33201 Section	Disposition
44	During shutdown conditions, in preparation for refueling, both trains of RWCU/SDC are running while the unit is in either Mode 5 or Mode 6 until the reactor cavity is flooded.	16.3	Operational Program – Procedure Development
45	The outage planning and control program is consistent with NUMARC 91-06.	16.3	Operational Program – Procedure Development
46	The FAPCS vessel injection manual isolation valve is a locked-open valve. While its open position is assured by administrative controls, it is an important valve whose failure to remain open could disable two active low pressure injection functions: FAPCS and FPS through FAPCS.	4.7	Operational Program – Human Factors Engineering
47	The PCCS pool drain line maintenance valves are locked-closed manual valves with position indication in the Main Control Room.	4.19	Operational Program – Human Factors Engineering
48	A fire in the lower drywell that damages all equipment in the area can significantly impact the CDF. These fires have been screened from the Fire PRA analysis. The area is inert during power operations. During shutdown, the screening is based on engineering judgment. The components that lead to the high risk significance are RWCU/SDC equipment and containment isolation valves. The judgment to screen this from analysis is based on the physical separation of the components, the limited number of ignition sources in the area, and the limited combustible material in the area.	12.10	Operational Program – Site Baseline PRA

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 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
Initiating 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.

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
High Pressure Mitigation	HPCS/ HPCI RCIC CRD 2-Division Analog Actuation	Isolation Condenser CRD 4-Division Digital Actuation	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.
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.

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
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 Room 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.
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.

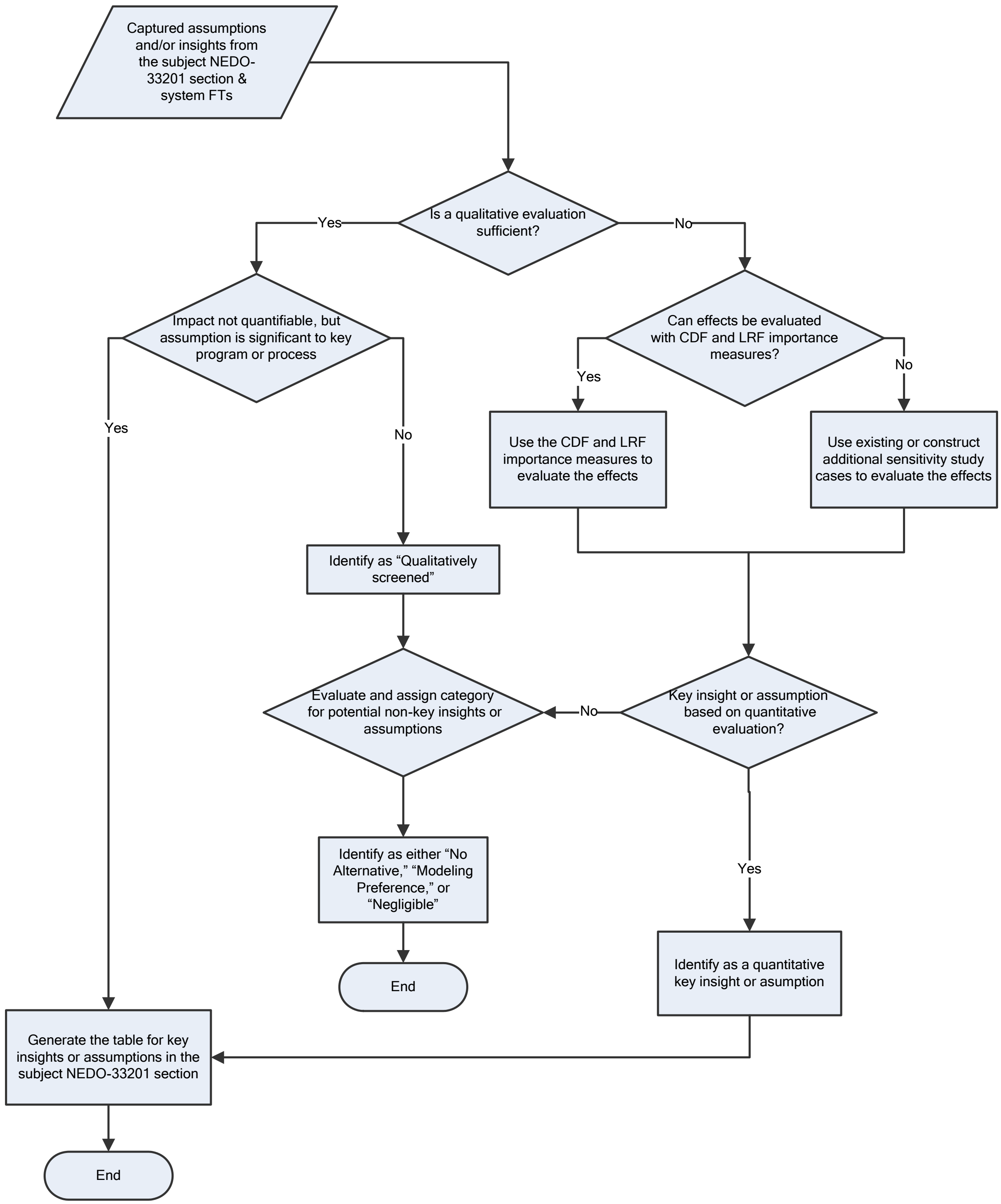

Figure 18-1. Process for Identification of Key Insights and Assumptions

Table 18-4

Analysis of Insights and Assumptions (DELETED)