19.4 External Event Analysis and Shutdown Risk Analysis

19.4.1 External Event Review

The Advanced Light Water Reactor (ALWR) Utility Requirements Document (Reference 19.4-1), contains a set of design requirements for the ALWR. The Nuclear Regulatory Commission's Severe Accident Policy (Reference 19.4-2) requires that a probabilistic risk assessment (PRA) be performed for any future nuclear power plant design and that it include consideration of potential external accident initiating events. Therefore, in order to provide a uniform basis for performing these required evaluations, a PRA Key Assumptions and Groundrules Document (Reference 19.4-3) is included as Appendix A of the ALWR Requirements Document. This appendix defines the purpose and scope of the PRA, as well as the types of events to be analyzed and those to be explicitly excluded.

The PRA Key Assumptions and Groundrules (KAG) Document explicitly addresses external initiating events and identifies those events that may be excluded based on qualitative evaluation, as well as those which may require quantitative assessment, for each ALWR. Potential external events identified in the PRA Procedures Guide (Reference 19.4-4) were considered to comprise an exhaustive listing of external events which should be considered for an ALWR PRA.

Potential external initiators identified for exclusion, as well as the accompanying rationale for exclusion, were reviewed. These included all identified events other than tornados and earthquakes. It is assumed that the EPRI assessment that the events listed are considered not to be important contributors to ALWR core damage based on improved design, proper siting, and low probability is an acceptable and sufficient basis for the exclusion of these events from more detailed evaluation in the ABWR PRA.

The above position is supported by the external event evaluations described and the conclusions drawn in Reference 19.4-5. This work was performed by ARSAP in support of the EPRI Requirements Document effort, and is judged applicable to the ABWR design. Assessments were made on the basis of the PRA Procedures Guide, siting requirements contained in the NRC Standard Review Plan (Reference 19.4-6), and EPRI ALWR design criteria.

Only two potential external accident initiators are identified by EPRI in the Key Assumptions and Groundrules Document as events which may require quantitative assessment for each ALWR:

(1) Tornado strikes

(2) Earthquakes

This EPRI assessment is the basis for limiting quantitative external event treatment to these two potential initiators. However, the NRC subsequently required additional analyses for internally initiated fire and flood. Treatment of all four initiators is discussed below.

EPRI qualitative assessment of the ALWR vulnerability to tornado-induced events concludes that most of the vulnerabilities found in past PRAs are not likely to occur in the ABWR design. Rather, the dominant effect of a tornado strike is expected to be prolonged loss of offsite power, and the EPRI position is that a simplified model is sufficient for assessment, provided that it addresses combinations of random failures in combination with loss of offsite power. The Advanced Reactor Severe Accident Program assisted EPRI by developing a method and model to quantitatively evaluate tornado strike impact. This EPRI KAG approach is applied in Subsection 19.4.2 to assess ABWR tornado vulnerability.

EPRI concludes that a seismic analysis is a required part of an ALWR PRA and presents the bases and rationale for its performance in the KAG document. The detailed seismic event analysis is presented in Subsection 19.4.3.

A screening analysis was performed for risk from internally initiated fires. This analysis was initially performed using the FIVE methodology and subsequently was replaced by a simplified fire PRA using the NUREG/CR-6850 Methodology. The results of the analysis are given in Subsection 19.4.4.

A probabilistic analysis was performed for flooding. All buildings which contain equipment that could be used for safe shutdown were considered. Subsection 19.4.5 contains the results of this analysis.

Shutdown risk was considered in order to evaluate the potential risk for Operational Modes 3, 4 and 5. The results of the analysis are given in Subsection 19.4.6.

19.4.2 Tornado Strike Analysis

As indicated in the preceding subsection, tornado strikes are one of the two classes of external initiating events which, according to the EPRI ALWR Utility Requirements Document, require quantitative assessment for each ALWR. This subsection discusses the basis for the EPRI position, describes the application of this high-level analytic approach to the ABWR, and presents results of the ABWR tornado strike evaluation. This EPRI position and approach are used to estimate ABWR tornado strike core damage frequency.

As part of the support provided EPRI in the development of the ALWR Requirements Document, ARSAP performed an evaluation of ALWR designs (as defined by the EPRI ALWR Requirements Document) to identify vulnerabilities to tornado events, and developed a model and approach to quantitatively estimate expected ALWR core damage frequency due to tornado strikes. Results of this activity are documented in Reference 19.4-7. The ARSAP qualitative evaluation indicates most vulnerabilities found in past PRAs are not expected to occur in the ABWR and that the dominant effect of a tornado strike is expected to be a prolonged loss of offsite power. Therefore, the need for analysis specifically addressing the consequences of tornado-induced loss of offsite power is indicated.

The ARSAP tornado evaluation developed expected tornado strike frequencies from regional historical data summarized in an EPRI report on tornado missile risk assessment (Reference 19.4-8). Tornados with intensities expected to contribute to core damage events were combined to generate total regional frequencies per square mile per year. Expected tornado strike frequency was then obtained by multiplying the regional values by an assumed plant area of approximately 0.363 square kilometers (0.14 square miles). The resulting regional site strike frequencies were found not to be strongly region dependent, and therefore the maximum assessed regional value was conservatively specified as the basis for evaluation.

Consequently, the loss of offsite power and station blackout accident event trees of Subsection 19D.4 were evaluated using the regional value as the loss of offsite power initiating event frequency in Figure 19D.4-4. In addition, these trees were adjusted to be consistent with the following assumptions resulting from the ARSAP qualitative evaluation of the expected ALWR tornado strike vulnerabilities:

- Condensate storage tank and condenser assumed vulnerable to tornado effects and no credit taken for either.
- Power conversion and feedwater systems assumed unavailable due to loss of offsite power.
- Offsite power recovery not expected within 24 hours following a tornado strike.

Remaining assumptions and conditions for evaluating the loss of offsite power and station blackout event trees for tornado site strike consequences were the same as those documented in Subsection 19D.4.

Evaluation of these event trees on the conservative bases listed above yields an extremely small total core damage frequency due to tornado-initiated events, which is quite small compared to the internal events result and the core damage frequency goal. Since tornado-induced events are predicted to be such small contributors to core damage frequency, this high level evaluation is judged to be sufficient and a more detailed analysis is not warranted.

19.4.3 Seismic Margins Analysis

19.4.3.1 Introduction

A seismic margins analysis (SMA) has been conducted for the ABWR using a modification of the Fragility Analysis method of Reference 19.4-9 to calculate high confidence low probability of failure (HCLPF) accelerations for important accident sequences and accident classes. HCLPF values were calculated for components and structures using the relationship

$$\mathrm{HCLPF} = A_m * \exp(-2.326 * \beta_c)$$

where:

A_m = the median peak ground acceleration corresponding to 50% failure probability,

β_c = the logarithmic standard deviation of the component or structure fragility.

The resulting HCLPF acceleration corresponds essentially to the 95th percent confidence level that at that acceleration the failure probability of a particular structure or component is less than 0.05 (5%). HCLPFs for accident sequences were evaluated through use of event trees, and seismic system analysis was performed with fault trees to determine HCLPFs of systems.

The seismic margins analysis evaluates the capability of the plant and equipment to withstand a large earthquake (2*SSE).

This subsection discusses the background, objectives, and general approach to the seismic margins analysis. The ground rules and analytical bases for the analysis are also given.

19.4.3.1.1 Background

Seismic event probabilistic analyses have been performed for several PRAs including the WASH-1400 Reactor Safety Study (Reference 19.4-10). The following statement was made in WASH-1400:

> "Although it is difficult to predict with precision the probability of potential accidents due to earthquake damage to a nuclear power plant because of general sparsity of quantitative data on the sizes and effects of earthquakes, it appears possible to make order of magnitude estimates that are useful in the type of risk assessment performed in this study."

Even though there has been a great deal of seismic research and analysis since WASH-1400, the above statement remains largely true today, particularly in regard to uncertainty in establishing an appropriate seismic hazard function. Because of the high degree of uncertainty that presently exists in this regard, a method of analysis has been developed that does not require prediction of an expected seismic hazard function. This methodology, a "seismic margins analysis", assesses the seismic capacity of the ABWR design in relation to the safe shutdown earthquake (SSE), and in relation to hypothetical seismically induced accident sequences that could lead to damage to the reactor core.

Section 3.2 states the following:

> "ABWR Standard Plant safety-related structures, systems, and components, including their foundations and supports, that are required to perform nuclear safety-related functions during or after a safe shutdown earthquake (SSE) are designated as Seismic Category I.

> "The Seismic Category I structures, systems and components are designed to withstand, without loss of function, the appropriate seismic loads (as discussed in Section 3.7) in combination with other appropriate loads."

Section 3.7 describes the deterministic analyses performed to verify the Standard Plant design relative to seismic events within the design basis envelope. Since the ABWR standard plant is designed for a nominal 0.3g SSE on all soil conditions described in Appendix 3A, considerable margin exists relative to any particular site. It is this design margin that allows the plant to accommodate seismic events far beyond the design basis without significant risk to the public health and safety. The seismic margins analysis discussed in the following subsections and presented in detail in Appendix 19I confirms the low risk for the ABWR standard plant from seismic-initiated events.

19.4.3.1.2 Objectives of the Analysis

The main objectives of the seismic margins analysis are the following:

(1) To provide assurance that the ABWR standard plant meets the intent of the NRC policy statement on severe accidents which includes consideration of seismic and other external events as requirements for plant certification.

(2) To provide insights and understanding of the relative contribution to seismic risk of the individual components and structures of the plant.

(3) To provide an understanding of the most probable sequences of events following a seismic event, and to identify any outstanding vulnerabilities (if any exist) to seismic events.

The seismic margins analysis that has been performed as described in the following subsections achieves the above objectives.

19.4.3.1.3 General Approach to the Analysis

The general approach and methods used in this analysis correspond to guidelines established by the NRC.

This assessment consists of five primary tasks:

(1) identification of critical structures, systems, and components (SSCs) in regard to potential seismically-initiated accident sequences,

(2) determination of the seismic capacity of critical components and structures,

(3) development of event tree models of potential seismic accident sequences,

(4) development of functional fault tree models of critical systems,

(5) assessment of the seismic margins (HCLPFs) of the ABWR in responding to the seismic accident sequences.

The first step in the analysis is to identify SSCs that are important to safety during a seismic accident and that may be vulnerable (to some extent) to seismic shock. In performing this step, use is made of the internal event analysis (Section 19.3 and Appendix 19D) and a general knowledge of component fragilities. The objective is to limit the size of the analysis by screening-out non-critical SSCs and SSCs that can obviously withstand a severe earthquake without functional damage.

The second step in the analysis is to determine the seismic capacity of the critical structures and components. Seismic capacities of generic components are based on past analyses. Seismic capacities of ABWR unique components are based on analysis or expected capability. The location of components in the plant configuration in relation to structures that may fail is also established. A structural fragility analysis is then conducted for all structures that contain important safety components. The component and structural fragilities are determined in terms of the median value of ground acceleration that would result in failure of the component or structure. Two additional parameters are derived defining the spread of the distribution about the median value.

The next step in the analysis is to construct event trees representing the potential seismically-induced accident sequences. In constructing the event trees and analyzing the event sequences, bases and assumptions of the analyses are established. Important bases of the analysis are listed in the following Subsection (19.4.3.1.4).

The next step in the analysis is to establish the seismic margins of the critical systems by constructing and analyzing functional fault tree models representing the systems.

The final step is to determine the HCLPFs of each of the identified potential accident sequences in the event trees. A computer program is used to perform this step.

19.4.3.1.4 Ground Rules/Analytical Bases

In addition to the PRA bases discussed in Section 19.2, several additional groundrules pertaining to the seismic analysis are given below:

(1) Because of the relatively low seismic capacity of ceramic insulators and offsite transmission lines, the analysis assumes that offsite power will be lost. No credit is given to recovery of offsite power when lost due to the seismic event. This may be somewhat conservative, but is necessary due to the uncertainty of the nature of the failure and actions necessary to recover power.

(2) No credit is given to repair or recovery of mechanical failure of components caused by the seismic event.

(3) Structural failure of a Seismic Category I building containing important equipment results in functional failure of all contained equipment.

(4) Seismic failure of identical redundant components at similar locations are treated as dependent failures i.e., all components fail together. This conservative assumption is used to simplify the analysis. At some future time, it may be desirable to selectively modify this assumption to provide a more accurate model.

19.4.3.2 Seismic Capacity Analysis

In order to determine the capacity of the ABWR plant to resist seismic events, it is necessary to know the seismic margins of plant structures and components. This is accomplished through the development of fragility curves and associated HCLPF capacities. A typical fragility curve is an S-shaped curve which has an increasing probability of failure at higher seismic motion. The mean fragility curve is given in terms of the mean peak ground acceleration (PGA). Fragility curves are generated for those components and structures that have been identified as potentially important to the seismic risk analysis. The resulting HCLPF capacities serve as input to the system analysis following the seismic margins approach.

The development of the seismic capacities for the structures and components of interest is given in Appendix 19H.

19.4.3.2.1 Structural Fragility

Detailed fragility evaluations are presented in Subsection 19H.3 for the following Seismic Category I structures:

- Reactor building shear walls
- Containment
- Reactor pressure vessel pedestal
- Control building.

19.4.3.2.2 Component Fragility

Seismic fragilities of safety-related components were assessed for the following two categories of components:

(1) ABWR Specific Components—whose fragility evaluation was made according to existing design information.

Detailed seismic fragility evaluations are presented in Subsection 19H.4.2 for the following ABWR specific components:

- Reactor pressure vessel (RPV),

- Shroud support,
- Control rod drive (CRD) guide tubes,
- CRD housings,
- Fuel assemblies.

(2) Generic Components—whose fragilities are based on data recommended in Reference 19.4-3 or other data sources as appropriate.

Detailed fragility evaluations for safety-related components other than those specific components presented above cannot be made at this stage of certification due to lack of design details.

The ABWR generic components of interest for this seismic risk analysis are the following:

- Off-site Power (transformers and ceramic insulators)
- Cable trays
- Batteries and battery tracks
- Battery chargers/Inverters
- Electric equipment (chatter failure mode)
- Switchgear/Motor control centers
- Transformers (480 V)
- Diesel generators and support systems
- Turbine-driven pumps
- Motor-driven pumps
- Diesel-driven pumps
- Heat exchangers
- Small tanks (e.g., standby liquid control tank)
- Air-operated valves
- Motor-operated valves

- Safety relief, manual, and check valves
- Hydraulic control units
- Large flat-bottom storage tanks
- Heating, ventilation, and air conditioning ducting
- Air handling units/room air conditioners
- Piping
- Service water pump house

These generic seismic capacities are selected from a review of ALWR recommendations (Reference 19.4-3) and other PRA studies. They are considered achievable for the ABWRs with an evolutionary improvement in the seismic capacities of the components designed to a 0.3g SSE.

19.4.3.3 Evaluation of Seismic Margin

The HCLPFs of accident sequences due to seismic events were calculated by constructing and quantifying event trees and fault trees which model the logical relationship of components, systems, functions, and structures that are significant to seismic risk. While structures were not modeled in the internal events analysis of Section 19.3, their inclusion is necessary in the seismic analysis because of the potential for component or function failure due to structural failure. In this analysis, it was assumed that all components housed within a failed structure would fail to function.

Fault trees and event trees are quantified to determine HCLPFs of systems and accident sequences. There are two alternative methods of quantification—“convolution” and “min-max”. In the convolution method, accident sequences are evaluated by combining input fragility curves according to the Boolean expression for each sequence. Seismic and random/human failure probabilities are calculated and combined (convolved) for discrete intervals of ground acceleration, and then integrated over the range of interest.

In the min-max method, input fragilities are combined by using the lowest (minimum) HCLPF value of a group of inputs operating in an OR logic, and by using the highest (maximum) HCLPF value of a group of inputs operating in an AND logic. Random/human failure probabilities are reported in combination with HCLPFs for each accident sequence.

Analysis of the effects beyond core damage (Level 2 PRA analysis) was not a part of this seismic margins analysis. However, event trees were constructed to examine the possibility of loss of containment isolation resulting in a large release given the earthquake and a resulting core damaging accident.

Because of the inclusion of a rupture disk in the ABWR design as an ultimate means of containment heat removal, and because an earthquake would not prevent rupture of the disk, failure of containment heat removal is not modeled in the seismic margins analysis. (There are no Class II sequences in the analysis.) There are two valves in line with the rupture disk; however, these valves are left in an open position, and the earthquake would not cause these valves to close.

19.4.3.4 Results of the Analysis

The convolution analysis is discussed in Appendix 19I. The HCLPF values for all accident sequences are greater than 0.60g, which is twice the safe shutdown earthquake (SSE = 0.30g).

For the accident sequences obtained from the min-max analysis, no accident sequence has a HCLPF lower than 0.60g.

For most accident sequences, the min-max method of analysis provided lower (more conservative) HCLPF values. However, the use of either method of analysis produced HCLPFs greater than twice the safe shutdown earthquake for all potential accident sequences. The seismic margins analysis has provided confidence that the ABWR design will withstand an earthquake of at least 0.6g intensity—twice the design SSE—and achieve safe shutdown without damage to the reactor core.

19.4.4 Fire Protection Probabilistic Risk Assessment

A fire PRA analysis was performed using NUREG/CR-6850 (Reference 19.4-15) methodology. This is a detailed fire PRA methodology.

For the ABWR design certification, a conservative approach to the NUREG/CR-6850 methodology was applied because plant specific design details, such as cable layout, are not available. Using the Level 1 PRA model and factoring in the fire-induced failures, core damage frequencies were evaluated and determined to be acceptable. A detailed fire probabilistic risk assessment is provided in Appendix 19M.

19.4.5 ABWR Probabilistic Flooding Analysis

The results of the ABWR Probabilistic Internal Flooding Analysis show that the turbine, control, and reactor buildings and the Reactor Service Water pump house , are the only structures that required evaluations for potential flooding. The other building do not contain any equipment that could be used for safe shutdown or potential flooding would not result in a plant transient.

Flooding in the turbine building could result in a turbine trip due to loss of circulating water or feedwater. Automatic pump trips and valve closure on high water level should terminate the flooding. But if these were to fail, a non-watertight door at grade level in the turbine building should allow water to exit the building. If this door retained water, watertight doors would

prevent water entering the control and reactor buildings. The core damage frequency (CDF) for turbine building flooding is extremely small.

The worst case flood in the control building is a break in the reactor service water system (RSW) which is an unlimited source. Floor drains and other openings in the floor would direct all flood water to the first floor where the reactor component cooling water (RCCW) rooms are located. The RCCW rooms contain sump pumps. Water level sensors in the RCCW rooms should actuate alarms in the control room and send signals to trip the RSW pumps and close isolation valves in the RSW system. If these sensors were to fail, watertight doors on each room should limit flood damage to only one of the three RCCW divisions. Breaks in the fire water system could result in interdivisional flooding in the upper floors but floor drains would limit water height to below installed equipment for the first hour. To prevent damage to safety-related equipment after this time requires operator actions to limit the depth of water. The CDF for control building flooding is extremely small.

Reactor building flooding could occur either inside or outside secondary containment. In either case, the flooding sources are finite with the suppression pool and condensate storage tank being the largest sources. Inside secondary containment flooding cannot cause damage to equipment in more than one of the three safety divisions on the first floor because of watertight doors on each safety division room. As was the case in the control building, water from breaks in lines on upper floors will be directed by floor drains to sump pumps on the first floor. The available volume of rooms on the first floor can contain all potential flood sources. Outside secondary containment, floor drains direct all flood water to sump pumps on floor B1F (third floor). If the sump pumps fail or cannot keep up with the flooding rate, an overfill line in the sumps direct water to the corridor of the first floor where it can be contained as discussed above. Interdivisional flooding may occur but floor drains will limit the water elevation such that no damage to safety equipment will occur. The CDF for reactor building flooding is extremely small.

The RSW pump house is contiguous with the Ultimate Heat Sink (UHS), and is separated into three divisions each with two levels, the pump room at the upper elevation, and the Electrical and HVAC room at the lower elevation. Each division is separated from the other divisions by watertight, three-hour fire rated doors at both elevations (RSW Interface Requirements, Subsection 2.11.9 (2)). For each RSW division, the RSW supply line from the Ultimate Heat Sink (UHS) splits in the RSW pump house to provide water to both RSW pumps. The RSW pump discharge combines into a single supply line to the Control Building. RSW return from the Control Building passes through the RSW pump rooms and returns to the UHS above the normal operating level. Flooding in the RSW pump house could occur from failure of the supply line from the UHS, or from failure in the RSW return line to the UHS. The RSW pump rooms contain sump pumps and water level sensors that function to mitigate the effects of small breaks or leaks from the RSW lines in the RSW pump room. If the sensors were to fail, watertight doors on each room and level should limit flood damage to only one of the three RSW divisions. Large breaks in the supply lines from the UHS are unisolable before the pump discharge isolation motor-operated valve (MOV). Breaks after the pump discharge MOV and in the RSW return line to the UHS are isolable with the pump discharge MOV, which closes automatically on high level in the RSW pump room. Unisolable breaks in the RSW supply piping will result in draining the UHS through the ventilation intake ducts in the top of the RSW

pump house. Breaks in the fire water system could result in interdivisional flooding in the upper floors of the RSW pump house, but floor drains would limit water height to below installed equipment for the first hour. To prevent damage to safety-related equipment after this time requires operator actions to limit the depth of water. The CDF for internal flooding in the RSW pump house is very small.

The total CDF for internal flooding is very small.

19.4.6 ABWR Shutdown Risk

The ABWR design has been evaluated for risks associated with shutdown conditions (i.e., Modes 3, 4 and 5). The evaluation included the following shutdown risk categories:

(1) Decay heat removal,

(2) Inventory control,

(3) Reactivity control,

(4) Electrical power (as a subset of inventory control and decay heat removal).

The evaluation also included risk reduction features of the ABWR due to instrumentation, flooding and fire protection, use of freeze seals, and procedure guidelines. ABWR features that are not part of current BWR designs were evaluated to determine if any new vulnerabilities would be introduced. In addition, an evaluation of approximately 200 events at operating BWR plants which were considered precursors to loss of decay heat removal capability showed that ABWR design features could mitigate the effects of all these events.

The results of the ABWR shutdown risk analysis demonstrated that the core damage frequency (CDF) for all shutdown event is very small. The main features that contribute to this low CDF are:

(1) Three physically and electrically independent residual heat removal (RHR) and support systems.

(2) Multiple makeup sources for inventory control (e.g., suppression pool, condensate storage tank, AC independent water addition system).

(3) Two independent off-site sources of electric power and four on-site sources (three emergency diesel generators and a combustion turbine generator).

(4) Reactor protection system (RPS) and standby liquid control system (for boron addition) and interlocks, to prevent accidental reactivity excursions.

19.4.7 References

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