19.2 Introduction

This section provides background and defines the objective, scope, bases and methodology of the internal events ABWR PRA provided in Appendix 19D, 19E and 19F. It explains how the analysis was conducted and the analytical bases for the methodologies employed.

19.2.1 Definitions

In this study the following definitions are used for the assessment of core damage and risk, subject to the employed methodology:

(1) Probabilistic Risk Assessment

Probabilistic Risk Assessment (PRA) is defined as the systematic identification, analysis and calculation of the probabilities and consequences of occurrence of postulated accident sequences.

(2) Frequency of Core Damage

The probability of core damage during a given year of operation is approximated by the assessed frequency of core damage. The assessed frequency of core damage per reactor year is defined as the product of the expected frequency of initiating events per year and the estimated mean probability of core damage given the initiating events.

(3) Risk

Risk to the public is expressed in terms of the assessed average consequences per reactor year which is defined as the product of the assessed frequency of release categories per reactor year and the estimated average consequences per release category summed over all release categories.

19.2.2 Objective and Scope

The objective of this PRA is to assess the probability of core damage and risk associated with the ABWR as defined in earlier chapters of Tier 2. This is accomplished by evaluating the frequency and consequence of postulated accident sequences.

The PRA analyzes the ABWR at an average site as defined by the site-related assumptions in Subsection 19E.3. Except for the shutdown risk studies, the analysis assumes that the plant is at full power prior to the initiation of an accident. The risk associated with fuel handling, storage and waste disposal accidents are judged to be insignificant and are not evaluated.

19.2.3 PRA Basis

To the extent practical, the analysis has been performed on a realistic basis. Equipment capability, success criteria, and event sequences are modeled realistically to determine, as accurately as possible, the expected course of events and conditions. Wherever possible, major conservatisms were avoided and best estimates were made of the physical effects, phenomena or probability.

In 2010, an update of the original design certification PRA was performed to incorporate Toshiba ABWR design features and to estimate the impact of a more modern PRA treatment. This updated PRA included revised initiating event frequencies and component failure data, a new assessment of the most important non-THERP (Technique of Human Error Rate Prediction)-based human actions, updated common cause failure treatment, and new Level 1 internal events analysis and associated Level 2 analysis probabilities.

Changes in the ABWR design since the original design certification PRA was performed that are discussed in other sections of this document have been evaluated for their potential impact on the PRA and their need to be modeled in the update. Based on this review, only the change in the Ultimate Heat Sink conceptual design to use a fan-forced cooling tower was concluded to need to be included explicitly in the 2010 PRA model. Other changes were not implemented in the updated PRA model since it was judged that they would either have no effect due to the modeling used in the PRA (e.g., elimination of hydrogen recombiners), or would result in an improved (reduced) CDF (e.g., provision of two 100% CUW pumps).

In 2012, additional enhancements were made to the PRA models, which include a state-of-the-art treatment of operator action dependency in the Level 1 full power internal events PRA. The enhancements were subsequently carried over to shutdown PRA. A simplified fire PRA using the NUREG/CR 6850 methodology (Reference 19.2-10) was performed to replace the analysis using the FIVE (Fire Induced Vulnerability Evaluation) methodology.

19.2.3.1 Key Assumptions and Ground Rules

All of the plant system design detail which is usually required to complete a PRA was not available at the time of the original design certification PRA study for the approved DCD, and it was agreed to list key PRA assumptions. These assumptions are those which relate to systems which are outside the scope of Tier 2 or information about the detailed design which was not yet available for the original design certification PRA. However, the assumptions made during the original design certification PRA for the DCD due to design details being unavailable remain conservative for the current design and were used for the updated PRA. These assumptions form interface requirements or information for the COL applicant. A summary list of these assumptions is shown on Table 19.2-1 which also includes reference to the subsection in which the assumption is discussed in more detail and reference to a “confirming” subsection in which the applicant is advised to confirm the assumption.

Assumptions which were needed to conduct analytical studies are not included in the table, but are discussed in the appropriate section describing the study.

19.2.3.2 Failure Probability and Field Experience

Realistic component failure probabilities were extracted from domestic operating BWR experience and supplemented by generic component failure probabilities (Subsection 19D.3). The expected loss of offsite power frequency is taken from an NRC compilation of losses of offsite power at U.S. nuclear power plants for all years 1986 through 2004 (Reference 19.2-3). Equipment maintenance or test unavailabilities used in the initial ABWR PRA submittal were taken from the GESSAR II PRA and were based upon BWR experience. In subsequent discussions with NRC regarding applicability of the GESSAR II values to ABWR, it was agreed that ABWR T&M unavailabilities would be increased over those of GESSAR to provide utility operational flexibility. Consequently, T&M values for RCIC, HPCFB, HPCFC, RHRA, RHRB, and RHRC were each raised to 2% in the PRA model as shown in Table 19D.3-2.

19.2.3.3 Initiating Accident Events

The expected frequency of transient events is based upon operating BWR experience and is a conservative value based upon current industry data in NUREG/CR-6928 (Reference 19.2-8). The expected manual shutdown frequency of one per year is based upon a 1985 analysis of operating plant data (Reference 19.2-4). LOCA initiation frequencies are from NUREG/CR-6928 (Reference 19.2-8) and are based upon NUREG/CR-1829 (Reference 19.2-9).

19.2.3.4 System Interactions and Common Cause Failures

Five factors are considered and explicitly incorporated in the analysis of system interactions and common cause failures:

(1) Component commonality at the system level, such as a common initiating signal.

(2) Common divisional services such as common electric power buses or common service water loops.

(3) System dependency, such as ADS dependency on the operability of at least one of the five (two high pressure and three low pressure) emergency core cooling system pumps.

(4) Past experience of losing onsite or offsite power.

(5) Human errors.

19.2.3.5 Human Reliability

The probability of human error is incorporated throughout the analysis by explicit inclusion in the fault trees and event trees of Subsections 19D.6 and 19D.4, respectively. Two types of errors have been considered:

(1) Errors resulting from operator failure to act as directed by normal or emergency procedures.

(2) Errors that contribute to component failure to perform as intended because the component has not been properly calibrated or restored to its operational state as required by plant procedures. Additional discussion regarding human error prediction and its application in the ABWR PRA is provided in Subsection 19D.7. In general, human errors are expected to be minimized by operator training and symptom-oriented emergency procedures.

Updated human reliability analysis (HRA) values were obtained for events of high importance. This is documented in Subsection 19D.7.

19.2.3.6 Reliability Model Definitions

In the event tree analyses, all systems capable of RPV water makeup injection or containment heat removal are modeled as governed by the success criteria (Subsection 19.3.1.3.1). To simplify the analysis, all degraded core sequences are conservatively treated as “binary” core damage sequences, i.e., no partially successful operations of NSSS or BOP systems are considered. Once core damage and fission product release is predicted in an accident sequence, no coolant injection system repair or recovery is considered in the accident event trees. In certain cases, credit for system recovery has been taken in the containment event trees. If adequate RPV water level has been maintained following accident initiation, online repair or recovery of containment heat removal, water injection, and diesel generator systems are modeled.

19.2.3.7 Initial and End Point Conditions

All of the accident sequences in this analysis except those in the shutdown risk assessment are assumed to be initiated with the plant in normal steady-state operation at 100 percent power. This is consistent with the approach taken in the GESSAR II PRA (Reference 19.2-5) and the WASH-1400 Reactor Safety Study (Reference 19.2-6). Consideration has been given to low power and shutdown in the shutdown risk assessment in Appendix 19L and 19Q.

The conditions of this analysis are the conditions applicable to a mid-life plant with a end-of-cycle core. This provides the widest and best degree of applicability to an operating ABWR. Other conditions of operation are taken as normal with nominal containment and suppression temperatures and pressures and stable external environmental conditions.

Each accident sequence analyzed is terminated in one of two conditions—core damage or safe shutdown. Sequences terminating with a damaged core are then analyzed through the containment event trees. These accident sequences in containment event trees terminate with either successful core melt arrest and therefore no radioactive release, or release to the environment. The criteria for preventing core damage are defined in Subsection 19.3.1.3.1. Recovery or mitigation of core damaging events is investigated and included where appropriate.

For those sequences terminating in safe shutdown, the success criteria as defined in Subsection 19.3.1.3.1 are met. The accident sequence is taken to a point where the reactor is in a condition of hot stable shutdown with the mode switch in shutdown, the reactor subcritical, pressures and temperatures stabilized and within limits, containment and suppression pool cooling being maintained, and vessel water level controlled. The analysis is not carried to cold shutdown due to the potentially long time involved, the low power level and slow progression of events, and the wide variety of test, maintenance, operating, shutdown, or recovery actions that could be involved.

For otherwise successful sequences where suppression pool cooling is not available and the containment overpressure protection system operates to relieve pressure, the time available for recovery actions is extended to the degree that a wide variety of recovery actions are possible. Such accident scenarios are not evaluated further.

19.2.3.8 Source Term and Core Melt Phenomenology

Source term analysis and core damage phenomenology are analyzed as discussed in Appendix 19E. These analyses cover events and conditions depicted by the accident and containment event trees in Subsections 19D.4 and 19D.5.

19.2.4 Methodology

Methodology used in the ABWR PRA is consistent with the approach and procedures applied in the GESSAR II Probabilistic Risk Assessment, but utilizes current methods for computing the frequency of core damage. Radioactive release resulting from postulated accident sequences was performed using the MAAP code. A summary description and illustration of the basic procedure followed as well as definitions of the major tasks of the analysis are provided in Subsection 19D.2.1.

19.2.4.1 Outline of the Analysis

As illustrated in Figure 19D.2-1, the basic analysis procedure followed consists of four major sequential tasks:

(1) Assessing the frequency of core damage,

(2) Determining the frequency of fission product release from the core and from the containment,

(3) Calculating the quantity of fission products released, and

(4) Determining the consequences of radioactive release.

Procedures for performing these tasks are diagrammed in Figures 19D.2-2 through 19D.2-5. The first two tasks, which provide the input necessary to determine the magnitude and consequences of release, were performed in 2010 and later updated in 2012. These tasks are discussed in Subsection 19D.2. Procedures for assessing the quantities of fission products released are discussed in Subsection 19E.2 and the process for evaluating the consequences of radioactive release is addressed in Subsection 19E.3.

19.2.4.2 Fault Tree–Event Tree Analysis

Given an initiating event, probabilities associated with the accident sequences were evaluated in fault tree and event tree logic models. Approaches taken and methods used to construct and evaluate these models are discussed in Subsection 19D.2.3.

19.2.4.3 Containment Analysis and Key Results

Probabilistic evaluation of containment failure is based on a detailed analysis of the ABWR. The containment ultimate strength under postulated severe accident conditions is evaluated in Appendix 19F. Melt progression analysis is contained in Subsection 19E.2.

The pressure capability of the concrete shell of the prototypical design at ambient temperature is 1.342 MPa for the top slab region as determined by analysis. The pressure capability is estimated to be as high as 1.342 MPa for the cylindrical wall also determined by analysis. The thermal effect of the representative severe accident temperature of 533 K (500°F) on the pressure capability of the concrete shell is expected to be insignificant.

The pressure capability of the drywell head is governed by plastic yield of the torispherical dome. The plastic yield limit pressure is evaluated using an approximate formula developed by Shield and Drucker based on the upper and lower bound theorems of limit analysis. The median limit pressure is 1.025 MPa at 533 K (500°F).

The ultimate pressure capability of the containment structure is therefore limited by the drywell head. When the limit pressure is reached, the containment is conservatively assumed to depressurize rapidly.

Leakage through fixed (mechanical and electrical) penetrations is negligible compared to leakage through large operable penetrations such as the drywell head, equipment hatches, and personnel airlocks. The leakage potential for those operable penetrations was evaluated. Very small [less than 0.0127 cm (0.005 in.)] separation displacements of the sealing surfaces at 0.722

MPa were calculated for the pressure-unseating drywell head closure and equipment hatches. No significant leakage is therefore anticipated before the capability pressure is reached. However, for the purpose of source term calculations, leakage in terms of leak areas is conservatively estimated, assuming no sealing action from degraded seals at temperature above 533 K (500°F), for pressures below the COPS rupture pressure as:

Pressure	Leak Area	
MPa	Cm^2	In^2
0.100	0	0.00
0.412 [design]	0	0.00
0.460 [SIT]	0	0.00
0.515	7.9	1.23
0.584	17.9	2.77
0.653	27.8	4.31
0.722 [COPS setpoint]	37.7	5.85

At and below the structural integrity test (SIT) pressure of 0.460 MPa (52 psig), leakage is within the design limit and the equivalent leak area is negligible.

The evaluation of the accident progression was performed using the MAAP code as described in Subsection 19E.2. MAAP is an integrated code which considers all the important aspects of a postulated severe accident, including both in-vessel and ex-vessel phenomena. In order to accurately model the important phenomena for the ABWR, MAAP was used to run several different accident scenarios. Additional analyses were performed using separate effects models as described in Subsection 19E.2.

Inputs of the MAAP code are the plant parameters and event sequence information. Based on this information, MAAP provides information about the pressure and temperature loads on the containment during a postulated severe accident as well as determining the timing and magnitude of any fission product release given the structural containment performance.

19.2.4.4 External Consequence Analysis

Evaluation of external consequences is performed using the CRAC-2 computer code. This evaluation involves:

- Amount and type of fission product release.

- Behavior of the fission products after release from the plant.
- Effects on the population exposed to the fission products.

Input data for the CRAC analysis include containment release data, weather data, demographic data, health physics data, and evacuation assumptions. Details of the CRAC code calculations are provided in Subsection 19E.3.

The calculation of accident consequences starts with the postulated release of fission products to the environment. Following the postulated release, the computer code calculates hourly dispersion, cloud depletion, and ground contamination concurrently with population evacuation. Using the resulting air and ground contamination along with population location with respect to the moving plume and dosimetric models based on the health physics data, individual radiological doses are calculated in terms of early and latent exposure for populations within a 40 km (25 mile) radius of the site. From these exposures, risk is characterized in terms of individual risk of early fatality and injury, societal risk of increased cancer incidence, and probability of dose versus distance.

19.2.4.5 Consequence Analysis Results

Previous PRAs have used a CCDF or Complimentary Cumulative Distribution Function for presentation of results and as a method for comparisons to WASH-1400 and other PRAs. Such a curve provides a highly graphical representation of potential consequences as a function of probability but is extremely dependent upon site characteristics such as evacuation planning and correlation of weather statistics to population demographics which have not been developed for a standard plant site by the NRC. The consequence analysis is described in Subsection 19E.3.

19.2.5 References

19.2-1 Not Used

19.2-2 Not Used.

19.2-3 Eide, S.A, Gentillon, C.D., Wierman, T.E., et al. Reevaluation of Station Blackout Risk at Nuclear Power Plants: Analysis of Loss of Offsite Power Events: 1986-2004, Idaho National Engineering Laboratory, NUREG/CR-6890, December 2005.

19.2-4 “Development of Transient Initiating Event Frequencies for Use in Probabilistic Risk Assessment”, NUREG/CR-3862, Idaho National Engineering Laboratory, May 1985.

19.2-5 “GESSAR II, 238 Nuclear Island, BWR/6 Standard Plant Probabilistic Risk Assessment”, Appendix A, 22A7007, General Electric Company, March 1982.

19.2-6 "Reactor Safety Study, An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants", WASH-1400, NUREG-75/014, United States Atomic Energy Commission, October 1975.

19.2-7 Not Used.

19.2-8 Eide, S.A., Wierman, T.E., Gentillon, C.D., et al. Industry-Average Performance for Components and Initiating Events at U.S. Commercial Nuclear Powerplants, Idaho National Engineering Laboratory, NUREG/CR-6928, February 2007.

19.2-9 Tregonig, R., Abramson, L., and Scott, P., Estimating LOCA Frequencies Through the Elicitation Process, USNRC, NUREG/CR-1829 (draft), June 2005.

19.2-10 "*EPRI/NRC-RES Fire PRA Methodology for Nuclear Power Facilities*," Electric Power Research Institute (EPRI), and U.S. Nuclear Regulatory Commission, Office of Nuclear Regulatory Research (RES), EPRI TR-1011989 and NUREG/CR-6850, September 2005.

Table 19.2-1 Key PRA Assumptions

Summary Assumptions	Reference Subsection	Confirming Subsection
Condensate Storage Pool Volume	19E.2.1.2.1(1)	19.9.9
Battery Loading Profiles for Station Blackout	19D.4.2.9, 19E.2.1.2.1(5)	19.9.9
RCIC Room Temperature Less Than Equipment Design Temperature	19D.4.2.9, 19E.2.1.2.1(6)	19.9.9
Control Room Temperature Less Than Equipment Design Temperature	19E.2.1.2.1(7)	19.9.9
Reactor Service Water System Definition	19D.6.4.2	19.9.26
ECCS Test and Surveillance Intervals	19.3.1.6	19.9.13
Seismic Margins	19H.5	19.9.4