19.6 Measurement Against Goals

This section summarizes the goals established in the ABWR Licensing Review Bases (Reference 19.6-1) which relate to the prevention or mitigation of severe accidents. In each case the means by which the goal is satisfied is briefly identified, with references to other parts of this chapter which provide additional details.

19.6.1 Goals

The goals summarized in Table 19.6-1 were identified in the Licensing Review Bases. These goals are addressed in Subsections 19.6.2 to 19.6.9.

19.6.2 Prevention of Core Damage

From the internal events analysis (Subsection 19D.5.12.2), the core damage frequency was calculated. For external events, conservative, bounding analyses were performed which conclude that the total core damage frequency is less than the goal of 1.0E-5 and the goal is satisfied.

19.6.3 Prevention of Early Containment Failure For Dominant Accident Sequences

Two modes of early containment failure were identified.

(1) There was judged to be a small chance of drywell failure for sequences in which the core melts with the reactor vessel at high pressure. These are the sequences with EH (for “early” containment failure and “high” release) as the last two characters in the sequence designators. Since depressurization is very reliable and since failure is unlikely even if depressurization fails, the frequency of these failures is calculated to be negligible.

(2) Suppression pool bypass was also identified as being potentially risk significant. If the vacuum breakers fail open, the steam and fission products will not be retained in the suppression pool. Therefore, significant early releases may be possible. The probability of these sequences is insignificant.

A third category of events have been conservatively included in this group. ATWS events with successful core cooling but failure of reactivity control may lead to early containment failure due to high power levels. No containment event trees were developed for these sequences due to the very low accident class frequency. Therefore, an extremely small frequency from the Level 1 analysis is included.

The total frequency is negligible and a very small percentage of the total core damage frequency, so it is concluded that this goal is satisfied.

19.6.4 Hydrogen from 100% of Active Zirconium

A separate effects calculation (Subsection 19E.2.3.2) indicates that the containment can withstand the static pressure of about 0.618 MPa that would be generated were this maximum hydrogen production to occur. This is substantially below the Service Level C Limit of 0.77 MPa for the containment.

19.6.5 Reliable Heat Removal to Reduce Probability of Containment Failure

Containment heat removal capacity is addressed in detail in Subsection 19.3.1.3.1(b), in which success criteria are developed to show the systems necessary to prevent overpressurizing the containment. There are four general means of removing heat from the containment: normal heat removal through the main condenser, reactor water cleanup system, drywell cooling system, and residual heat removal system. No credit is given for operation of the drywell coolers in the PRA.

The heat removal path to the main condenser can be used when the main steamlines are open (or reopened).

The reactor water cleanup system can remove decay heat which is generated four hours after scram if the reactor vessel is at high pressure.

The Residual Heat Removal (RHR) System is described in Subsection 5.4.7. For the worst case event with scram, overpressurization of the containment can be prevented by any one of three RHR divisional subsystems (Loops A, B and C). These subsystems are located in different quadrants of the plant and are protected from common mode failures by divisional separation criteria. Each of these subsystems is automatically initiated by two-out-of-four logic for each initiating parameter (low reactor water level or high drywell pressure). Each of the four input signals to the two-out-of-four logic is provided by separate Class 1E instrument divisions. The heat exchanger is always in the cooling flow path, so that containment cooling to the ultimate heat sink starts as soon as RHR injection flow begins. Under normal operating conditions, there is one Reactor Building Cooling Water (RCW) pump, one Service Water (RSW) pump and two RCW heat exchanger in operation in each of the three loops. When required, the standby pumps and heat exchangers are put in operation. This cooling occurs for all modes of RHR operation. Any one of the following four RHR operating modes will satisfactorily prevent overpressurizing the containment for the dominant sequences:

(1) Low pressure flooder mode

(2) The suppression pool cooling mode

(3) The shutdown cooling mode

(4) The containment spray mode

Key components of the supporting RCW and service water systems (one pump and two heat exchangers in each loop as noted above) are in operation during normal plant operation. At least each month, the standby pumps and heat exchangers are started and the previously running RCW and RSW equipment is placed in a standby mode. This design and operating philosophy results in high reliability since, for example, if power is lost and later regained, the equipment in half of each loop can be thought of as having been tested within the last few minutes or few hours; the other half can be thought of as having been tested within the past week (Summer operation) or month (Winter operation). The RHR and supporting systems are also used to maintain low suppression pool temperatures during normal operation. Depending on the ultimate heat sink temperature, this could occur with frequencies ranging from about once per day to once per week.

The reliability of RHR and its supporting systems is assessed in the fault trees of Subsection 19.D.6.

Finally, as noted in Subsection 19.D.5.2 Accident Classes, (2) Class II, there is a substantial amount of time available during which the above heat removal systems might be repaired if they had initially failed. This time effectively increases the reliability of these systems.

The calculated frequency of containment structural failure resulting from loss of heat removal for internal events is extremely small (Subsection 19D.5.12). Only an extremely small percentage of these events result in core damage due to the high degree of diversity and redundancy in the core cooling systems.

Thus, very reliable heat removal is provided which makes the probability of containment failure very small and the goal is satisfied.

19.6.6 Prevention of Hydrogen Deflagration and Detonation

The inerting system is described in Subsection 6.2.5. The primary containment vessel is inerted with nitrogen gas to below 3.5% oxygen by volume. This provides some margin against instrumentation errors or an (unexplained) increase in oxygen concentration. Any increase in actual oxygen concentration is highly improbable since measures are taken to eliminate any source of oxygen in the containment. This includes substituting nitrogen for air in all pneumatic systems and seals, and maintaining the containment at a slightly positive pressure during reactor power operation to prevent in-leakage of air (oxygen). The containment oxygen concentration is expected (and has been observed in operating plants) to slowly decrease during prolonged power operation as nitrogen makeup is periodically added to compensate for the slight leakage from the containment at positive pressure. Therefore, the margin against entering a potentially flammable regime is normally more than 1%.

The ABWR follows standard industry inerting design, establishing a nitrogen storage and delivery system sufficient to inert the containment in less than four hours. In addition, deinerting (to at least 18% oxygen) is possible within four hours.

In the ABWR, the drywell cooler flow rate is very high such that the residence time (drywell volume divided by the inerting or deinerting flow rate) is on the order of a few minutes. Therefore, good mixing is assured. In addition to conserving nitrogen, this good mixing assures that “pockets” of uninerted atmosphere are swept away and the containment is truly inert. Pocketing in the wetwell is much less of a concern since it is a relatively open space.

In conclusion, the inerted containment atmosphere provides passive protection against hydrogen deflagration and detonation. This protection is not vulnerable to loss of power and is available during all accident sequences. Therefore, the goal is satisfied.

19.6.7 Offsite Dose/Large Release

The probability of a 0.25 Sv whole body dose at 0.8 km (1/2 mile) from the reactor is extremely small less than the 10^{-6} per year goal. This goal is satisfied. No attempt was made to define the term “large release” but the 0.25 Sv dose is considered to be “much less than large”, so the large release goal is satisfied.

19.6.8 Containment Conditional Failure Probability

A conditional containment failure probability was determined as outlined below.

19.6.8.1 Potential Mechanisms of Containment Failure

There are several potential mechanisms which could cause significant fission product release and thus might be considered to be “containment failure”:

(1) Energetic steam explosions.

(2) Hydrogen deflagration/detonation.

(3) Suppression pool bypass.

(4) High pressure/temperature combinations.

Energetic explosions are reviewed in Subsection 19E.2.3.1 and it is concluded that there is no potential for steam explosions of sufficient magnitude to overpressurize the containment. Without such overpressurization, there is no potential for significant fission product release. Therefore, for purposes of measuring against the goal, the probability of containment failure resulting from steam explosions is taken as zero.

Hydrogen deflagration/detonation is precluded by inerting the containment as discussed in Subsection 19.6.6. For purposes of measuring against this goal, the probability of containment failure resulting from hydrogen deflagration and detonation is also taken as zero.

Suppression pool bypass which results from certain random equipment failures before or during the accident (as opposed to bypass which results because of increasing temperature or pressure) are examined in Subsection 19E.2.3.3. For internal events, this evaluation showed that the conditional probability of full bypass is extremely small and that the contribution of this bypass is a small percentage of the total plant risk. The potential for suppression pool bypass was also considered from the standpoint of external events and no significant additional mechanisms were identified. Since the conditional probability of full bypass is much less than the 0.1 goal, this potential containment failure mechanism is not considered further for the purposes of measurement against this goal.

High pressure/temperature combinations within the containment under certain conditions can cause containment failure. These potential failures are treated in Subsection 19.6.8.2.

19.6.8.2 Definition of ”Containment Failure“

Containment failure is defined here in a manner which provides an indication of failure of the containment function: Containment failure is considered to have occurred for any sequence which gives an offsite dose at 0.8 km (1/2 mile) of 0.25 Sv or more. In general, this occurs as a result of increased pressure and/or temperature as noted below.

Increased leakage from the containment could occur through penetrations as a result of increasing pressure and temperature. Analysis in Appendix 19F indicates that this could occur if the containment pressure exceeds 0.46 MPa and the temperature exceeds 533 K (500°F).

Drywell head failure is most likely to occur as a result of reduced drywell head load carrying capability at increased temperatures as noted in Appendix 19F. Failure pressure is estimated at 1.025 MPa at an upper drywell temperature of 533 K (500°F).

19.6.8.3 Measurement Against the Goal

The frequency of exceeding 0.25 Sv at 0.8 km (1/2 mile) is extremely small. Dividing by the core damage frequency, gives a conditional containment failure probability less than the goal of 0.1 and the goal is satisfied.

Measurement against an alternate definition of containment failure based on maintenance of containment integrity is discussed in Subsection 19.5.3.

19.6.9 Safety Goal Policy Statement

The calculated individual risk is insignificant and the societal risk is negligible. The calculated risks are many decades below the numerical goals and the goals are satisfied.

19.6.10 Not Used

19.6.11 Conclusion

As noted in the discussions in Subsections 19.6.1 through 19.6.9, the Licensing Review Bases goals are satisfied.

19.6.12 References

19.6-1 Thomas E. Murley (NRC) letter to Ricardo Artigas (GE), August 7, 1987, "Advanced Boiling Water Reactor Licensing Review Bases."

Table 19.6-1 Summary of Goals in Licensing Review Bases

Licensing Review Bases Para. No.	Subsection In Which Goal Is Addressed	Summary Statement of Goal
7.5.1	19.6.2	**Prevention of Core Damage**—Mean core damage frequency from internal and external events less than 10^{-5} per reactor year.
7.5.2		**Mitigation of Core Damage**
7.5.2a	19.6.3	Measures to reduce the probability of early containment failure for dominant accident sequences.
7.5.2b	19.6.4	Measures to accommodate hydrogen generated from the reaction of 100% of the zirconium in the active fuel cladding.
7.5.2c	19.6.5	Highly reliable heat removal systems to reduce the probability of containment failure by loss of heat removal.
7.5.2d	19.6.6	Reliable means to prevent hydrogen deflagration and detonation.
7.5.3		**Offsite Consequences**
7.5.3(1)	19.6.7	Mean frequency of offsite doses in excess of 0.25 Sv beyond one-half mile radius (typical United States site boundary) from the reactor less than 10^{-6} per reactor year, considering both internal and external events.
7.5.3(2)	19.6.8	Containment design is to assure that the containment conditional failure probability is less than 0.1 when weighted over credible core damage sequences.
8.10		**Safety Goal Policy Statement**—Comply with eventual requirements. Since the eventual requirements are not known, the current policy is addressed here. The current policy is:
	19.6.9	Risk to average individual in the vicinity of the plant less than 0.1% of sum of prompt fatality risk from other accidents.
	19.6.9	Risk to population within 16.1 km of the plant less than 0.1% of sum of cancer fatality risk from other causes.
	19.6.7	In addition, although not stated in the Licensing Review Bases, Toshiba intends to satisfy the goal (Federal Register, page 28047) of overall mean frequency of large release less than 10^{-6} per reactor year. This goal is addressed along with the above Paragraph 7.5.3(1) goal relating to offsite doses.