GE Nuclear Energy

ABWR

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Subject New Sections 19.11 thru 19.13 of ABWR PRA

Message These sections (especially 19.13) should close the loop on punch list item 02213 ("PRA Insights")

cc JNFOX

19.11 HUMAN ACTION OVERVIEW

Late in the process of conducting this PRA, several functions, previously performed manually, were automated to reduce the dependence on human actions. The system fault trees were updated and the PRA results were requantified. In addition, other studies were performed to provide an improved understanding of human actions in the PRA.

Sensitivity studies of the core damage frequency resulting from the level 1 analysis were conducted (Subsection 19D.7.2). From this study, four human actions after accident initiation were found to be the most important. They are actions taken to provide water injection to the reactor vessel if the several automatic injection features fail to accomplish this function.

In addition, the PRA was reviewed to compile a list of human actions which were assumed in other parts of the analysis (Subsection 19D.7.3). From this list and the above mentioned sensitivity studies, actions were identified which should be given consideration as being "CRITICAL TASKS" as defined by the human factors evaluation Design Acceptance Criteria, as noted in Section 18E.2. These human factors are listed or referenced in Subsection 19D.7.4.

The human actions lists were also reviewed to ensure consistency with the ABWR emergency procedure guidelines (Appendix 18A). This review is documented in Appendix 18F. Some of the actions are not appropriate for inclusion in the symptom based emergency procedure guidelines. These are included in the COL applicant action item list in Section 19.9 "COL License Information."

19.12 PRA INPUT TO THE RELIABILITY ASSURANCE PROBLEM

The major results of the PRA were reviewed to determine the reliability and maintenance actions that should be considered by the COL applicant throughout the life of the plant. This review is documented in Appendix 19K.

The level 1 analysis results were reviewed by examining two important measures ("Fussell-Vesely" and "Risk Achievement Worth"). Individual systems and components were identified as being most important (Table 19K.3-1).

The balance of the PRA was reviewed (Sections 19K.4 through 19K.8) to determine other important features not addressed in the level 1 analysis.

The most important features thus identified were finally reviewed to determine appropriate maintenance and surveillance actions (Section 19K.9).

19.13 SUMMARY OF INSIGHTS GAINED FROM THE PRA

The PRA was conducted with several objectives in mind:

1) To ensure that the PRA-related goals in the ABWR Licensing Review Bases established in 1987 were satisfied.

2) To review and improve the design capability for potential weaknesses or relative vulnerabilities, not withstanding the achievement of the Licensing Review Bases goals.

3) To identify the most important aspects of the design and its operation so that particular attention can be placed in these aspects during certification, detailed design, plant operation.

4) To provide additional basic studies which were not anticipated when the Licensing Review Bases was established.

5) To provide uncertainty/sensitivity studies of key results.

The objectives were achieved as noted in the following subsections.

19.13.1 Licensing Review Bases Goals

These goals were established to ensure that an appropriate balance between accident prevention and accident mitigation is achieved by ABWR. The goals (Table 19.6-1 provides a summary) focus on prevention (core damage frequency less than 10^{-5} per year), mitigation (avoiding containment failure from several potential threats) and offsite consequences (as measured by offsite doses, consequences, conditional containment failure probability, and the Safety Goal Policy Statement).

Measurement against these goals and the features which are important in achieving the goals are discussed in detail in Section 19.6. (19.6 will later be revised in the SSAR to reflect new numbers.) The goals are satisfied, indicating a very robust design with an excellent balance between accident prevention and mitigation features.

19.13.2 The Search for Vulnerabilities

As noted in detail in Section 19.7, the PRA process was used extensively to improve the design, even though it could be argued that satisfying the goals of Section 19.6 was sufficient. Improvements were made in many areas, including for example: the automation of several accident prevention functions, the addition of a combustion turbine generator to improve power supply diversity, the addition of an ac independent water addition system to improve accident prevention and mitigation, and the addition of two passive accident mitigation features (the lower drywell flooder and the containment overpressure protection system) which substantially address uncertainties associated with severe accident progression. Procedural improvements were also identified. Many other examples are cited in Section 19.7 to illustrate the manner in which PRA techniques were used throughout the design process to improve the design.

19.13.3 The Most Important Aspects of the Design

The ABWR design and its operation was reviewed to determine the features and operator actions which are most important from a PRA perspective. Applying additional focus on these aspects can provide confidence that ABWR operation will be as accident resistant as characterized by the PRA.

The key design features were identified for input to the tier 1 design description, a key certification effort. These key features are provided in Section 19.8.

The potential for human error was reviewed extensively (Section 19.11) to ensure that "CRITICAL TASKS" were identified for the human factors Design Acceptance Criteria and to ensure that human actions are covered by the emergency procedures guidelines or other, more specific procedures.

The PRA results were reviewed to determine which surveillance and maintenance activities are most important throughout plant life (Section 19.12).

19.13.4 Additional Studies

Several additional studies which were not anticipated in the original Licensing Review Bases were conducted to further review the robustness of the ABWR design.

The potential for internal fires to lead to core damage is studied in Appendix 19M. The basic ABWR features of separating the three safety divisions into individual fire zones and the ability to control key systems from outside the control room are the major reasons that very low core damage frequencies are calculated.

Internal flooding is investigated in detail from both a deterministic and probabilistic perspective in Appendix 19R. Divisional and building separation along with other key flooding mitigation features are identified which lead to the conclusion that there is a very small threat posed by internal flooding. General guidelines for addressing the potential for severe external flooding are provided in Section 19.9.

A seismic margins analysis (Appendix reference later) was conducted to assess the potential for seismic events beyond the design basis to lead to core damage. It was determined that there is high confidence in a low failure probability, even at ground accelerations approximately two times the plant seismic design basis. Key components and their seismic capacities are identified so that the COL applicant can review the design capability against those assumed in this margins analysis.

An assessment of the potential for core damage to result from ABWR operations while shutdown is documented in Appendix 19Q. Potential precursor events are reviewed for their applicability to ABWR and several ABWR features are noted which reduce the risk from activities conducted while shutdown. A decay heat removal reliability study is conducted to provide input to the COL applicant as to which complements of decay heat removal and water addition system could be kept available while shutdown to reduce the risk of core damage resulting from the loss of an operating RHR system.

19.13.5 Uncertainty and Sensitivity Studies

After the plant system fault trees were updated to reflect several design improvements, the level 1 results were requantified. Then a data uncertainty study was performed (reference subsection later). The results show a mean core damage frequency of about 1.5E-7 events per year. The 95th percentile value is about three times this value, or 4.5E-7 per year. Thus, the effect of data uncertainty is relatively minor. The most important contribution to the uncertainty is the RCIC maintenance activity. This activity is addressed in the PRA input to reliability assurance (Appendix 19K).

A comparison of the requantified level 1 results to those for Grand Gulf was also developed (Due 7/15) to document the major reasons for reductions in the frequency of the various accident classes. The sensitivity of the results to equipment outage times and surveillance intervals was considered (Due 7/15). The contribution of human errors was compared to the contribution from an operating plant (Due date not yet established).

certainties associated with severe accident progression were examined in detail through the use of containment event trees supplemented by decomposition event trees. The later were used to study the potential for different outcomes of various severe accident events. The results show that the ABWR design is very robust. Analysis of phenomena such as direct containment heating were performed which indicate that the probability of occurrence with sufficient magnitude to fail the containment is very small. The design is not sensitive to assumptions affecting debris coolability due to its high strength and lower drywell/pedestal design. The studies also demonstrated that the features of the ABWR design substantially reduced the uncertainty associated with many severe accident phenomena. In many areas, these studies were conducted in greater depth than studies with similar objectives reported in NUREG-1150 and its supporting documents. In addition, the basis for the judgements made is described in detail.