19M Fire Protection Probabilistic Risk Assessment

19M.1 Introduction

As part of the Advanced Boiling Water Reactor (ABWR) design certification process, Toshiba has expanded upon earlier considerations of the subject of fire risk. In the ABWR original design certification, the Fire-Induced Vulnerability Evaluation (FIVE) methodology (Reference 19M-1) provided an appropriate vehicle for performing this analysis. In the years since the ABWR original design certification, the NRC and Electric Power Research Institute (EPRI) issued NUREG/CR-6850 (Reference 19M-2). The NUREG/CR-6850 fire probabilistic risk assessment (PRA) methodology is more detailed than the previous FIVE methodology. For the ABWR design certification renewal, the fire PRA has been updated by basing the analysis on the NUREG/CR-6850 methodology.

The fire PRA analysis is performed using more detailed methodology than the FIVE analysis but less detailed than a fire PRA that fully utilizes the NUREG/CR-6850 methodology. The level of detail and extent to which NUREG/CR-6850 is followed for the fire PRA analysis corresponds with the level of ABWR design documentation available at the time of analysis initiation. Since the details of the ABWR plant specific design and operating procedures are not currently available and plant walkdowns cannot be performed at this time, conservative assumptions are made. A bounding approach is used to demonstrate low risk for the purposes of design certification renewal.

19M.2 Basis of the Analysis

The technical tasks required to perform a fire PRA using the NUREG/CR-6850 methodology are shown in Figure 1 of Reference 19M-2. The tasks include procedures for identifying fire compartments for evaluation purposes, defining fire ignition frequencies, and performing quantitative screening analyses of fire risk. There are 16 tasks, which are described briefly in the following subsection.

In the FIVE methodology, the core damage frequency (CDF) results for each fire area is compared to a screening value. However, in the case of the fire PRA, CDF results for the whole plant are compared to the total plant CDF goals (Section 19.6).

Details of the fire PRA analysis are contained in the "ABWR Fire Probabilistic Risk Assessment," UTLR-0019 (Reference 19M-3).

19M.2.1 NUREG/CR-6850 Tasks

The 16 fire PRA NUREG/CR-6850 tasks are listed below.

Task 1: Plant Boundary Definition and Partitioning

Task 2: Fire PRA Component Selection

Task 3: Cable Selection

Task 4: Qualitative Screening

Task 5: Plant Fire-Induced Risk Model

Task 6: Fire Ignition Frequencies

Task 7: Quantitative Screening

Task 8: Scoping Fire Modeling

Task 9: Detailed Circuit Failure Analysis

Task 10: Circuit Failure Mode Likelihood Analysis

Task 11: Detailed Fire Modeling

Task 12: Post-Fire Human Reliability Analysis (HRA)

Task 13: Seismic Fire Interactions

Task 14: Fire Risk Quantification

Task 15: Uncertainty and Sensitivity Analyses

Task 16: Fire PRA Documentation (see Reference 19M-3)

With the completion of Task 1, which defines the fire compartments, and Task 4, which screens out the fire compartments having little or no contribution to fire risk, seven bounding fire compartments are developed, along with corresponding ignition frequency estimates (Task 6) for each compartment.

The first three compartments consider the impact of fires that incapacitate each of the three divisions of emergency power, and thus the ECCS equipment that is dependent on each for successful performance. These three safety-related compartments consist of the reactor building except primary containment, control building except the control room complex, and the intake structure. This grouping contains all of the equipment required for safe shutdown except that within the primary containment and the control room complex. Any fire in a divisional area is conservatively assumed to result in the immediate and complete loss of function of the division.

The fourth fire compartment considers the impact of a fire in the control room. The fifth compartment considers nonsafety-related fire areas that only contain nonsafety-related equipment that is separated from the other fire compartments. The sixth compartment examines the consequences of a fire in the turbine building. The seventh and final fire compartment considers the general yard area outside the buildings.

Task 2 selects plant components and failure modes to be modeled in the fire PRA. Included in Task 2 is the identification of fire risk multiple spurious operations (MSOs) that can potentially cause undesired component operation.

Task 5 develops the fire PRA model and quantification process for determining fire-induced conditional core damage probabilities (CCDPs).

For the Task 11 main control room (MCR) fire scenario evaluation, the contribution of the MCR fire scenarios to the plant CDF is estimated. A quantitative assessment of the MCR fire risk, including both abandonment and non-abandonment scenarios, is included in the analysis. For the abandonment scenario, it is assumed that the only ECCS functions available are those that can be controlled and operated from the remote shutdown panel, and the RCIC, which can be manually operated outside of the control room. For the non-abandonment scenario, it is assumed that only one control panel (or a portion of a larger control panel) is lost before the fire is extinguished, and therefore the plant can be brought to shutdown from inside the MCR.

The human reliability analysis (Task 12) evaluates the impact of fire scenarios on human actions by identifying human failure events (HFEs) to be included in the fire PRA model and then assigning human error probabilities (HEPs) for the identified HFEs. A simplified fire HRA process is used.

Task 14 performs the final quantification of the fire PRA results. The fire-induced CCDP model from Task 5 and the ignition frequency for each compartment from Task 6 are used to calculate the total fire-induced CDF. This then is refined, yielding the final CDF estimate.

In Task 15 sensitivity studies are performed to evaluate the potential impact of significant sources of uncertainty on CDF results. The CDF contribution of MSO was evaluated as part of the sensitivity studies.

Tasks 3, 7, 8, 9, 10, and 13 were not completed for the ABWR certification fire PRA analysis due to non-availability of ABWR detailed plant-specific design information. If plant-specific design details were available, the impact of fire in each compartment could be limited by carrying out these tasks. However, in the absence of detailed plant-specific design information, such as cable and equipment layout, all equipment in a fire compartment was assumed to burn and become unavailable as a result of a fire. Assuming full burn out results in conservative estimates of core damage frequency.

19M.3 Summary of Results

Because ABWR design details were not available, conservative assumptions were made in developing the PRA model. With this model, total fire PRA plant CDF values were evaluated to be significantly less than the PRA acceptance criteria (Section 19.6).

For detailed results of the fire PRA analysis performed, see Reference 19M-3.

19M.4 Not Used

19M.5 Not Used

19M.6 Not Used

19M.7 Not Used

19M.8 References

19M-1 "Fire Vulnerability Evaluation Methodology, FIVE, Plant Screening Guide", Electric Power Research Institute, Preliminary Draft.

19M-2 "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.

19M-3 UTLR-0019, "ABWR Fire Probabilistic Risk Assessment," Toshiba.

The following tables are not used in the DCD:

Tables 19M-1 through 19M-14

The following figures are not used in the DCD:

Figures 19M-1 through 19M-13