GE Nuclear Energy

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Docket No. STN 52-001

Chet Poslusny, Senior Project Manager
Standardization Project Directorate
Associate Directorate for Advanced Reactors
and License Renewal
Office of Nuclear Reactor Regulation

Subject: Submittal Supporting Accelerated ABWR Schedule - **Modelling Uncertainty in PRA Success Criteria**

Dear Chet:

Enclosed are SSAR markups addressing modelling uncertainty in PRA success criteria. These markups will be included in the next amendment.

Please provide a copy of this transmittal to Glenn Kelly.

Sincerely,

Jack Fox
Advanced Reactor Programs

cc: Alan Beard (GE)
Norman Fletcher (DOE)
Joe Quirk (GE)
Frank Paradiso (GE)
Jack Duncan (GE)

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A motor driven feedwater pump is combined in series with a condensate pump in order to provide a higher pressure system. Therefore, this option also depends on the availability of makeup water and electrical power. Sufficient makeup water is available to enable this series of pumps to maintain adequate core cooling for the small steam LOCA and transient events.

The fire protection system has two pumps which take suction from the firewater tanks and inject into the RPV through an RHR line. One pump is driven by an electric motor which requires AC power. The other is driven directly by a diesel engine. Once the reactor system has been depressurized, either pump can provide enough makeup water to restore and maintain the RPV water level following any transient (including IORV) event. The analysis to support this conclusion assumes a full ADS blowdown begins within 15 minutes after the vessel water level has reached the level 1 setpoint. The subsequent reactor system depressurization allows injection from the fire protection system about 7 minutes after the start of the blowdown. The ability of the fire protection system to mitigate the consequences of LOCA events is conservatively ignored. For more information about the fire protection system refer to Subsection 5.4.7.

INSERT A

(b) Containment Heat Removal

Following the success of the core cooling function, heat must be removed from the containment. Containment heat removal is considered a success if the containment pressure is kept below the pressure at which loss of containment integrity is estimated to occur (Appendix 19F). Successful containment heat removal can be achieved by using the RHR System or, depending on the circumstances as defined in Table 19.3-2, the normal heat removal path or the CUW System. The resultant ABWR longterm heat removal success criteria to prevent initial core damage for transient and Loss of Coolant Accident (LOCA) events with RPS scram are given in Table 19.3-2.

The RHR has four major modes of operation and heat is removed from the containment in each of these modes. During the core cooling mode which is initiated automatically, the RHR heat exchanger is in the loop and the heat removal process is established. If core cooling is accomplished without the use of an RHR System, and the suppression pool begins overheating, the suppression pool cooling mode of the RHR will be automatically or manually initiated by the operator. Once initiated, an RHR System will begin removing heat from the containment and eventually terminate the pool heatup.

Insert A

It is conservative to use the 2200°F PCT licensing limit as an acceptance criteria for the success criteria since tests have been performed which show that the core will remain in a coolable geometry with temperatures as high as 2700°F.

A review of Table 19.3-2 shows that, for success, the inventory threatening events require the flow equivalent of only 1 RHR/LPFL or 1 HPCF pump available for large break cases and only 1 HPCF or 1 RHR/LPFL + 3 ADS available for small break cases. The resulting PCTs for the large break cases and transients were between 900°F and 1100°F. For the small break cases with the flow equivalent of only 1 HPCF available the resulting PCTs were less than 1000°F and with 1 RHR/LPFL + 3 ADS available the maximum PCT was 1800°F.

Subsection 6.3.3.7.8 identifies the input parameters that significantly impact the LOCA results. If the above analyses were reanalyzed with these conservative input parameters, it is estimated that only the resulting PCTs for the small break cases with 1 RHR/LPFL + 3 ADS available are above 1800°F. For these cases the PCT is estimated to be about 2300°F. However, even for these convervative LOCA calculations all the PCTs are less than 2700°F which is still acceptable and most LOCA cases and transients are much less than 2700°F. Therefore, there is no need to include an uncertainty analysis in the generation of the success criteria.

19D.10 Data Uncertainty for ABWR PRA

19D.10.1 Introduction

This analysis presents the results of a quantitative data uncertainty analysis for the Advanced Boiling Water Reactor (ABWR) Level 1 Probabilistic Risk Assessment (PRA). ~~Neither modeling uncertainty nor~~ completeness uncertainty ~~were~~ was not analyzed. Modelling uncertainty is addressed in subsection 19.3.1.3(2)

19D.10.2 Purpose and Summary of Conclusions

The purpose of this study was to determine and propagate data uncertainty in the internal events analysis in the ABWR Level 1 PRA, to provide the probability distribution describing the uncertainty in the calculated core damage frequency (CDF).

Subsequent to the performance of this study, a more detailed assessment of Level 1 PRA importance measures revealed that the importance of the combustion turbine and each diesel generator had been understated. Results of these later analyses are presented in Table 19K-1 and show theses components to rank number one and two, respectively, in Fussell-Vesely importance. These greater values of importance are not included in the uncertainty analyses which follow; and, these components would be expected to rank at the top of the list of contributors to uncertainty in CDF presented in Table 19D.10-5. It is expected that revision of these analyses to reflect incorporation of these higher values would not materially affect the conclusions.

The uncertainty analysis results show that the ABWR CDF has the distribution shown in Figure 19D.10-1, having a mean value of 1.56E-07 per reactor-year and an error factor of 4.2, (calculated as the 95th percentile divided by the median). The 95th percentile of the distribution is 2.9 times the mean value or 4.53E-07. The 5th percentile is 3.4E-08 per reactor-year.

The top ten contributors to the uncertainty in the CDF were identified using the uncertainty importance measure. Nine of these are also in the top ten basic events ranked according to the Fussell-Vesely (F-V) importance measure. The basic event RCIMAINT (i.e., RCIC is down for test or maintenance) is the highest contributor to uncertainty in the CDF as well as to the mean value of the CDF. RCIC test and maintenance is part of the reliability assurance program (RAP), and is discussed in Subsection 19K.9. The remaining contributors are identified in Subsection 19D.10.6.1.

The results of the uncertainty show that the 95th percentile is only moderately sensitive to the error factors (EFs) of the basic events, and hence that lack of precise EF values has a rather small effect on the outcome. For example, doubling the EF values of each basic event simultaneously increases the 95th percentile of the CDF by only 12%. When all EFs are set equal to 15, the 95th percentile increases by only 14%. (Note 1 in Subsection 19D.10.8).