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Page 8, Table 2:

* Diesel Mission Time was increased from 2.5 to 14 hours to account for the increased time expected to recuver offsite power derived from data analysis published in NUREG/CR-5496.

Page 9, Table 3.a:

NOTE 1: To simplify the data analysis, the analyst assumed that the ratio of high and low pressure sequences were the same as for internal events baseline. This has been accepted practice for achieving a reasonable approximation for ΔLERF.

Page 11, First paragraph:

As described in the IPEEE, the licensee determined that there were three different potential fire scenarios in the service water pump room, namely: a fire damaging one pump, caused by a small oil-spill fire limited to a 2-quart spill from the lower bearing reservoir associated with that pump; a fire that results from the spill of all the oil from a single pump (28 quarts), spreading rapidly, and damaging three pumps; and fires that affect all four pumps. The licensee had determined that fires affecting only two pumps were not likely, because of the nature of oil spills and spreading calculations. The analyst determined that a four-pump fire was part of the baseline risk, therefore, it would not be evaluated. A one-pump fire would not automatically result in a plant transient. However, the analyst assumed that a three-pump fire affecting both of the Division I pumps, would result in a loss of service water system initiating event.

Page 14, Several paragraphs:

In accordance with Manual Chapter 0609, Appendix H, "Containment Integrity SDP," the analyst determined that this was a Type A finding, because the finding affected the plant core damage frequency. The analyst evaluated both the baseline model and the current case model to determine the LERF potential sequences and segregate them into the categories provided in Appendix H, Table 5.2, "Phase 2 Assessment Factors - Type A Findings at Full Power. The primary distinctions in categories are based on the initiator type, the pressure of the reactor coolant system at the time of core damage, and whether the drywell floor has been flooded, either by the event or by operator action. The type of event is indicative of the mode of core damage and the available systems; the coolant system pressure indicates whether the core will melt through or be ejected from the vessel; and in a Mark I containment, the steel line is significantly more susceptible to melt through if there is no water on the drywell floor. The categories, the total core damage frequency related to each of these categorizations, the LERF factors, and an estimation of the change in LERF are documented in Table 5 of this worksheet.

Following each model run, the analyst segregated the core damage sequences as follows:

- Loss of coolant accidents were assumed to result in a wet drywell floor. The analyst assumed that during all station blackout initiating events the drywell floor remained dry. The Cooper Nuclear emergency operating procedures require drywell flooding if reactor vessel level can not be restored. Therefore, the analysts assumed that containment flooding was successful for all high pressure transients and those low pressure transients that had the residual heat removal system available.

- All individual intersystem loss of coolant accident initiators designated in the SPAR model were grouped in the ISLOCA category.

Page 16, First Bullet:

- As stated in Assumption i in the above analysis, the analyst used a value of 0.4 for the probability that operators would fail to realign gland water prior to failure of the Division II pumps. This value was derived using the INEEL's SPAR-H method. The licensee used a Human Error Probability of 9.2×10^{-2} derived from for the probability that operators would fail to realign gland water prior to failure of the Division II pumps. The analyst used a failure probability of 0.4, derived from the INEEL's SPAR-H method.