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SUBJECT: Special rept:in Feb 1990 anomaly observed between results of new large break LOCA analysis & analysis of record.Caused by error in RELAP4 input deck.Mod re use of measured nominal loop flow rate instituted.

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H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2
DOCKET NO. 50-261/LICENSE NO. DPR-23
SPECIAL REPORT
LARGE BREAK LOSS OF COOLANT ACCIDENT ANALYSIS

Gentlemen:

Carolina Power & Light Company provides this Special Report as required by 10CFR50.46 and as a follow-up to the preliminary notification made via the Emergency Notification System at 5:11 p.m. on October 23, 1990. The fuel vendor, Advanced Nuclear Fuels Corporation, has completed a reanalysis of the large break loss of coolant accident with the results as stated below.

<u>SUMMARY</u>

In February 1990, an anomaly was observed between the results of a new large break loss of coolant accident analysis and the analysis of record. An evaluation was immediately initiated to determine the nature of the anomaly. The reason for the anomaly was determined to be related to data transfers between the RELAP4 system blowdown calculation and the RELAP4 hot channel calculation. The anomaly was subsequently determined to constitute an error in the RELAP4 input deck. An evaluation of the significance and impact of the error was then undertaken. The result of the evaluation was that the error was significant, as defined by 10CFR50.46(a)(3)(i), in that the peak cladding temperature increased more than 50°F (from 1982° to 2178°F), but is still within the acceptance criteria of 10CFR50.46(b).

<u>EVALUATION</u>

The analysis of record was performed by Advanced Nuclear Fuels Corporation in 1987 on a Control Data Corporation computer. When the system and hot channel blowdown calculations were repeated on the current CRAY computer, as part of the new analysis, it was observed that the fuel rod temperatures at the end of blowdown were higher than in the analysis of record. A large number of sensitivity calculations were performed to identify the reason for the anomaly. The anomaly was finally determined to be related to the frequency of data being stored on the RELAP4 system blowdown tape, which is subsequently used for boundary conditions in the hot channel calculation. The boundary condition data is interpolated in the hot channel calculation to recalculate core flow rates. The frequency of data stored on the RELAP4 system boundary condition tape is determined through user input by the combination of the maximum time step size allowed (sec) and the number of time steps specified per edit on the tape.

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Sensitivity studies on boundary conditions edit frequency indicated that the edit frequency used in the analysis of record was too coarse for the hot channel calculation to adequately reproduce core flow rates predicted in the RELAP4 system calculation. The edit frequency used in the current evaluation appropriately allowed the hot channel calculation to reproduce the core flow rates predicted in the RELAP4 system calculation, consistent with the sensitivity studies.

In addition to the change in edit frequency, a review of input data characterizing the H. B. Robinson Plant was conducted. Two modifications were made to the analysis to better reflect current H. B. Robinson Plant operation. These modifications are consistent with Advanced Nuclear Fuels' NRC-approved methods. The modifications incorporated into the analysis were (1) the use of the measured nominal loop flow rate (corrected to a steam generator tube plugging level of 6 percent) rather than the design flow rate and (2) the use of more realistic radial pin-to-pin power distribution. In addition, more realistic combinations of fuel stored energy and axial power shape as a function of burnup were used in the analysis. This last change had little impact on the peak cladding temperature.

Break spectrum calculations were rerun through end-of-bypass time in the hot channel runs to verify that the 0.8 Double-Ended-Cold-Leg-Guillotine break size remains limiting. These calculations were followed by stored energy and axial shape sensitivity calculations to determine the significance of the error and to verify that 10CFR50.46(b) criteria are met with current Technical Specification values. The limiting case was determined to be the case with the combination of Beginning-of-Cycle stored energy and a Middle-of-Cycle axial power shape, peaked at 73 percent of core height. The peak cladding temperature was calculated to be 2178°F.

Advanced Nuclear Fuels has evaluated other Pressurized Water Reactors' analyses and determined that this type of input error does not affect other Pressurized Water Reactor plants supported by Advanced Nuclear Fuels' large break loss of coolant accident analyses. RELAP4 edit frequencies were found to be acceptable for all other plants. Advanced Nuclear Fuels' corrective actions did include the implementation of RELAP4 edit frequencies shown by sensitivity studies to be acceptable for this aspect of loss of coolant accident analyses.

CONCLUSION

In conclusion, an anomaly was identified in the analysis of record. This anomaly was investigated; the cause of the anomaly was identified and corrected. The large break loss of coolant accident analysis has been rerun, and the results verified that the criteria of 10CFR50.46(b) are satisfied. The reanalysis did show a peak cladding temperature increase of 196°F and is therefore being reported as required by 10CFR50.46a(3)(ii).

Should you have any questions regarding this report, please contact Mr. R. W. Prunty at (919) 546-7318.

Yours very truly,

S. D. Floyd

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Manager
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