UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING ACCEPTANCE OF CONFIRMATORY ECCS SMALL BREAK ANALYSIS

AMENDMENT NO. 25 TO FACILITY OPERATING LICENSE NO. DPR-3

YANKEE NUCLEAR POWER STATION (YANKEE-ROWE)

DOCKET NO. 50-29

Introduction

By application dated May 11, 1976, and supplements dated May 11, 17, 19, 1976, and June 16, 1976, Yankee Atomic Electric Company (the licensee) proposed a change to the Technical Specifications appended to License No. DPR-3 for the Yankee-Rowe reactor. The proposal involved a change of the set pressure for the safety injection accumulator, specified in Section 3.2.e(4), from 410 psig to 337 psig ±10 psig. The staff's safety evaluation, dated May 19, 1976, concluded that adjusting the accumulator pressure to match previous ECCS calculations was acceptable for large break LOCAs. Although the analyses submitted in May 1976 were adequate to demonstrate that the revised accumulator pressure resulted in ECCS performance in conformance with the Commission's Acceptance Criteria for large breaks and for small breaks of 4-inch diameter or less, these initial analyses did not provide computations for small breaks larger than 4-inch diameter. The licensee asserted that the larger breaks (e.g. 5-inches) would result in faster depressurization and more rapid injection of emergency core coolant which would result in more rapid recovering of the core. The Staff's May 19, 1976 safety evaluation concluded that the 5-inch break is likely to be less limiting than the 4-inch break. The Staff accepted the licensee's correction of accumulator pressure on an interim bases, provided that the licensee submitted improved small break LOCA analysis to confirm that the most limiting small breaks had been identified and conformed to the Commission's Acceptance Criteria.

The licensee submitted such analyses on June 16, 1976. This Safety Evaluation describes the Staff's evaluation of this confirmatory analysis.

Staff Evaluation

Prior to the June 16, 1976 submittal, small break analyses were performed by the licensee using the WFLASH code and submitted to the NRC on July 31, 1974. These analyses considered 2, 3 and 4-inch diameter breaks. Results of these analyses indicated that the core never became uncovered for the 2" break. A peak clad temperature (PCT) of 750°F was calculated for the 3-inch break and the 4-inch break was determined to be limiting with a PCT of 1300°F occurring 235 seconds after break initiation.

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The 4-inch break was re-analyzed using the ENC WREM based evaluation model. Results of this analysis were included in the licensee's October 10, 1975 application for a license amendment. The calculated PCT during the period of core recovery was determined to be 1400°F at 230 seconds. However, a PCT of about 1850°F was calculated to occur at about 9 seconds due to a momentary flow stagnation. Small breaks larger than 4-inch diameter had not been previously conducted using the SLAP code in 1971 which indicates that the core never uncovered.

The licensee's confirmatory small break analyses were conducted with the approved RELAP-4-EM SMALL BREAK code (RELAP4-EM/003 11/07/75 95 ENC20). For fuel heatup calculations, the approved TOODEE2 ENC13 Code was used. Small breaks of 2.00, 2.25, 4.0, 5.0 and 10 inch ID were analyzed. Corrected accumulator pressure and flow resistances were used. Direct spillage of ECCS fluid from the safety injection line in the broken loop to the containment floor was assumed to occur for the 2.25 inch ID and all larger breaks. The 2.25 inch ID break corresponds to a complete severence of a safety injection line. No direct spillage of ECCS fluid to the containment was assumed to occur for breaks smaller than 2.25-inches ID. In these cases, the reactor coolant system (RCS) pressure is felt in each loop of the ECCS system including both pumps and the accumulator.

The peak clad temperature analysis was performed using the TOODEE2 digital computer code. For the 4, 5 and 10-inch ID breaks, the peak power fuel rod was divided into 11 axial nodes. Initial fuel rod temperature distributions were obtained from a steady-state analysis using a RELAP4-EM/HOT CHANNEL model with a corresponding axial nodalization. Time dependent fluid conditions required as TOODEE2 input came from the blowdown results.

For the 2.0 and 2.25-inch ID breaks, in which the upper portion of the core was uncovered for a longer time period relative to the rest of the core, a 19 axial node model was used to simulate the power rod. This permitted a detailed representation of the upper portion of the rod which experiences a relatively long heatup period.

The limiting small break LOCA was that resulting from a safety injection (2.25-inch ID) pipe rupture for which the peak clad temperature was calculated to be 1872°F, occurring at 1170.2 seconds. Previous analyses had shown the 4.0-inch ID pipe rupture to be limiting.

The reason for the shift in the limiting small LOCA break size was due to the fact that a smaller break size, the 2.25-inch ID safety injection pipe rupture, was identified in which direct spillage of ECCS fluid could occur. This break results in a slower RCS depressurization such that ECCS injection into the intact loops occurs much later in the accident that in the 4-inch break, thereby allowing a substantially longer heatup period.

For the 4, 5 and 10-inch ID breaks, depressurization was fast enough to allow the ECCS system to inject into the RCS sooner and to terminate the transient. For breaks smaller than 2.25-inches ID (size of the thermal sleeve in the ECCS penetration into the RCS), no direct spillage produced more effective ECCS performance.

Conclusions

From its review of the June 16, 1976, confirmatory small break analysis, the Staff concludes: (1) the analysis acceptably covers the small break spectrum and identifies the limiting small break size, (2) the correct ECCS accumulator parameters (pressure and flow) were used in the calculations, and (3) the analysis methods and results are acceptable and do not involve a decrease in the safety margin for the Yankee-Rowe Core XII ECCS performance evaluation.*

Environmental Considerations

We have determined that the amendment does not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that the amendment involves an action which is insignificant from the standpoint of environmental impact and pursuant to 10 CFR §51.5(d)(4) that an environmental impact statement, or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of this amendment.

Conclusion

We have concluded, based on the considerations discussed above, that: (1) because the amendment does not involve a significant increase in the probability or consequences of accidents previously considered and does not involve a significant decrease in a safety margin, the amendment does not involve a significant hazards considerations, (2) there is reasonable

*Subsequent to the review reflected in this evaluation, we obtained new information requiring a modification of upper head water temperature assumed in the ECCS evaluation for Yankee-Rowe. As a result, the NRC on August 27, 1976 issued an Order for Modification of license, restricting the peak linear heat generation rate to 0.85 kw/ft, and requiring a revised ECCS evaluation.

This amendment correcting accumulator pressure, does not affect the requirement or conclusions of the Order of August 27, 1976. The only effect the Order has on considerations discussed in this safety evaluation would be to maintain or reduce the peak clad temperature set forth herein.

assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Date: October 7, 1976