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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 89 TO FACILITY OPERATING LICENSE NO. DPR-39

AND AMENDMENT NO. 79 TO FACILITY OPERATING LICENSE NO. DPR-48

COMMONWEALTH EDISON COMPANY

ZION NUCLEAR POWER STATION, UNITS 1 AND 2

DOCKET NOS. 50-295 AND 50-304

SUMMARY

A revised ECCS analysis was performed for Zion 1 and 2. Changes in the analysis assumptions included: 10% steam generator tube plugging (uniform distribution); 15x15 Optimized Fuel; a maximum accumulator water volume of 888 cubic feet; and a containment spray actuation time increase to 92.5 seconds. The analysis also assumed a 102% power level, and an increase in the total peaking factor (F_q) from 2.13 to 2.32 (Reference 1).

The analysis was performed with the revised 1981 Westinghouse ECCS Evaluation Model (Reference 2). This version includes the BART computer program, a mechanistic core heat transfer model (References 3 and 4). This constitutes an approved ECCS model.

The large-break LOCA cases analyzed were for a double-ended, pump discharge, cold leg break with discharge multipliers of 0.4, 0.6 and 0.8. The small-break LOCA analyzed was a 6-inch diameter cold leg break. These cases have been shown, in previous analyses, to be the limiting breaks with respect to peak cladding temperatures.

Supplemental information regarding the containment spray actuation time, the new containment fan cooler heat removal curves, and non-LOCA transients and accidents was also provided (Reference 5). The spray actuation time was set to 45 seconds, the original FSAR value. Also provided was an assessment of the impact of these changes on non-LOCA transients and accidents.

In addition to the revised analyses, the report (Reference 1) included changes to the Technical Specifications for Zion 1 and 2 to reflect the changes in accumulator water inventory and the peaking factor of 2.32 (F_q).

Additional information concerning asymmetric steam generator tube plugging, maximum permissible plugging, the non-LOCA design base calculation basis, and accumulator water volume for the small-break LOCA was requested from the licensee. The licensee has provided the requested information (Reference 6).

Subsequent to these submittals, Westinghouse notified the NRC of an error in the Evaluation Model methodology used to transfer WREFLOOD generated data into the BART computer program (Reference 7). The licensee has submitted a revised large break ECCS analysis for which the BART input more accurately reflects the time integrated volume of water delivered to the core based on the WREFLOOD output curve (data) for the core inlet flooding rate (Reference 8).

The revised ECCS analysis, based on the supplemental analysis (Reference 8), results in a peak cladding temperature of 2159 degrees Fahrenheit, total core hydrogen generation of less than 0.3%, and local cladding oxidation of 6.94% for the limiting break (0.6 discharge coefficient). The peak cladding

temperature for the small-break LOCA is 1747 degrees Fahrenheit, with core wide and local oxidation values of less than 0.3% and 1.45% respectively (Reference 1). These values are within the required limits as specified in 10 CFR 50.46.

STAFF EVALUATION

This revised ECCS performance analysis has been made using an NRC-approved evaluation model which satisfies the requirements of 10 CFR 50.46 and Appendix K to Part 50. Both large-break LOCAs and small break LOCAs have been considered. The methods for accounting for steam generator tube plugging are an integral part of the Westinghouse evaluation model and have been previously approved by the NRC. The difference in steam generator tube plugging levels, between the worst case steam generator and the best case steam generator, is currently less than 3%. Westinghouse has considered the effects of asymmetric tube plugging and has determined that asymmetric plugging levels which do not exceed the analyzed uniform plugging levels (e.g., 10% uniform plugging, maximum 10% difference between any steam generators) will have no adverse impact upon transients. Commonwealth Edison will not plug greater than 10% of the tubes in any steam generator or exceed more than 10% asymmetric plugging without further analysis and prior notification to the NRC (Reference 6).

The staff has reviewed the calculation of the containment backpressure in the revised ECCS analysis and determined that the analysis was performed using a previously approved model for containment parameters. The original (Reference 1) analysis was performed using a containment spray flow initiation time of 92.5 seconds. During the review process, the licensee, Commonwealth Edison, determined that a spray initiation time of 50 to 57 seconds, depending on the

offsite power availability assumptions, was more appropriate. The licensee has conducted a revised analysis in support of the Reactor Containment Fan Cooler requirements and provided a re-analysis for the ECCS evaluation (Reference 5). Changes in the revised analysis include: (1) a containment spray actuation time of 45 seconds (the original Zion FSAR value); (2) the new reactor containment fan cooler heat removal curves (an increased capacity); and (3) minor enhancements in the treatment of fuel pellet density and volumetric heat generation calculations in the LOCTA-IV initialization scheme (Reference 9).

A review of the input assumptions and parameter values used for the small-break LOCA analysis resulted in the identification of an inconsistency in the accumulator water volume used. The input and assumptions used for the small-break LOCA analysis, with the exception of the core linear power distribution, are consistent with the September 1974 10 CFR 50, Appendix K submittal, and the analysis was performed using the currently approved ECCS small-break models. The analysis is reported to use a 900 cubic foot volume, whereas the revised value as specified in the Technical Specifications requires a maximum volume not in excess of 888 cubic feet.

Since this appeared to be inconsistent with 10 CFR 50.36, which requires that Technical Specifications be derived from the safety analysis, the licensee was requested to justify how compliance was met. In response (Reference 6), the licensee has provided additional information for the limiting small-break LOCA. In May 1981, a new large-break LOCA was approved which featured a reduced water inventory in the accumulators from 900 cubic feet to 868 cubic feet. The small-break LOCA did not have to be reanalyzed on the basis of the accumulator

water inventory change. Unlike the large-break LOCA, the accumulator water volume change from 900 to 868 cubic feet is unimportant in mitigating Zion's small-break LOCA transient. Westinghouse determined that an accumulator water volume of 800 cubic feet was more than adequate to cover Zion's small-break LOCA. The core recovers very quickly in the Zion limiting case small-break LOCA analysis once the accumulators have actuated. Less than 300 cubic feet of water injected from each of the available accumulators recovers the core in the limiting break transient. The remaining water in the accumulator tanks provides the inventory to maintain the downcomer and core full until the system depressurizes to approximately 200 psia, at which time pump safety injection flow matches break flow. Continued flow from the accumulators is not needed to prevent another core uncovery with less than 600 cubic feet of water per accumulator tank having delivered. Therefore, both the current accumulator volume Technical Specification limit of 868 cubic feet and the proposed value of 888 cubic feet protect the core during small-break LOCA scenarios. The proposed Technical Specification increase to 888 cubic feet will provide consistency with the new analysis for the limiting large-break LOCA. The effects of these changes on all other transients or classes of accidents in addition to the LOCA have been evaluated by Westinghouse and the licensee (Reference 5). It was determined by Westinghouse and Commonwealth Edison Company that the Thermal Design Flow (TDF) and the maximum primary coolant average temperature (Tave) of the Zion Units bound the 10% steam generator tube plugging situation in the non-LOCA transient analyses. The non-LOCA transients are analyzed with a nuclear heat flux hot channel factor (F_q^N) of 2.32. The concerns regarding flow reduction due to 10% uniform tube plugging have been addressed by Westinghouse. The Thermal Design Flow (TDF) has been shown by analysis to be unaffected by 10% uniform tube plugging. The impact of the

reduced heat transfer area in the steam generators has also been analyzed and demonstrated to be negligible. No change in the Tave program is necessary nor will any change be affected. Therefore, since both the Thermal Design Flow and Tave program are unchanged, the margin of safety as calculated in the existing transient and accident analyses is unaffected for a 10% tube plugging level.

Compliance with 10 CFR 50.46 was demonstrated using previously approved LOCA Evaluation Models. For non-LOCA events, it was demonstrated that the current transient and accident analyses are unaffected by the 10% steam generator tube plugging level.

We note, however, that item 1A of the 1980 order requires that the PCT not exceed 2050°F using the October 22, 1979 model. Therefore the licensee should use the more limiting of either the peaking factor from the current analysis or the peaking factor from the analysis on which the 1980 Order is based.

STAFF FINDINGS

The staff has reviewed the calculation of the containment backpressure in the revised ECCS analysis and determined that the analysis was performed with a previously approved model and containment parameters. We find the revised analysis acceptable.

This revised ECCS performance evaluation has been made using an NRC approved evaluation model which satisfies the requirements of 10 CFR 50.46 and Appendix K to Part 50. A 10% uniform steam generator tube plugging is incorporated into the model. A maximum total peaking factor of 2.32 (F_q) was used resulting in a peak cladding temperature of 2159 degrees Fahrenheit for the limiting break.

The licensee also evaluated the potential effects of these changes on non-LOCA accidents and transients. It was concluded that the Thermal Design Flow will be met and that the maximum primary coolant average temperature assumed in previously approved analyses will not be exceeded. These changes will therefore not involve a reduction in the margin of safety with respect to DNB and other criteria for non-LOCA events.

The changes to the Technical Specifications are supported by the safety analyses provided by the licensee.

We find this evaluation acceptable, subject to the licensee's commitment not to exceed 10% steam generator tube plugging in any one steam generator or exceed more than 10% asymmetric plugging without further analysis and prior notification to the NRC. This commitment was docketed March 8, 1985.

We note that until the order item A.1 is lifted or amended, an F_q limit of 2.2 is necessary in order for the PCT to remain below 2050 using the October 22, 1979 model.

The licensee's submittal of March 8, 1985, made as a result of NRC staff request to clarify the original submittal, corrected a calculational error in the original submittal dated October 19, 1984.

Environmental Consideration

These amendments involve a change in the installation or use of the facilities components located within the restricted areas as defined in 10 CFR 20. The staff has determined that these amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that these amendments involve no significant hazards consideration and there has been no public comment on such finding. Accordingly, these amendments meets the eligibility criteria for categorical exclusion set forth in 10 CFR Sec 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of these amendments.

Conclusion

We have concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner,

and (2) such activities will be conducted in compliance with the Commission's regulations and the issuance of these amendments will not be inimical to the common defense and security or to the health and safety of the public.

Dated: May 24, 1985

Principal Contributor:

E. Throm

REFERENCES

1. Letter from R. N. Cascarano, Nuclear Licensing Administrator, Commonwealth Edison, to H. R. Denton, Director, Nuclear Regulatory Commission, Subject: Zion Generating Station Units 1 and 2 Revised ECCS Analysis and Proposed Amendment to Facility Operating License Nos. DPR-39 and DPR-48 NRC Docket Nos. 50-295 and 50-304, dated October 19, 1984.

2. Rahe, E.P., "Westinghouse ECCS Evaluation Model, 1981 Version," WCAP-9920-P-A Rev. 1 (Proprietary), WCAP-9221-P-1 Rev. 1 (Non-Proprietary), 1981.

3. Letter from D. G. McDonald (NRC) to R. E. Uhrig (FP&L) transmitting NRC's SER concerning WCAP-9561, "BART A-1, A Computer Code for Best Estimate Analysis of Reflood Transients," December 21, 1983.

4. Letter from D. G. McDonald (NRC) to R. E. Uhrig (FP&L) transmitting Turkey Point SER based on BART.

5. Letter from R. N. Cascarano, Nuclear Licensing Administrator, Commonwealth Edison, to H. R. Denton, Director, Nuclear Regulatory Commission, Subject: Zion Generating Station Units 1 and 2 Revised ECCS Analysis and Proposed Amendment to Facility Operating License Nos. DPR-39 and DPR-48 NRC Docket Nos. 50-295 and 50-304, dated February 14, 1985.

6. Letter from R. N. Cascarano, Nuclear Licensing Administrator, Commonwealth Edison, to H. R. Denton, Director, Nuclear Regulatory Commissiom, Subject: Zion Generating Station Units 1 and 2 Revised ECCS Analysis and Proposed Amendment to Facility Operating License Nos. DPR-39 and DPR-48 NRC Docket Nos. 50-295 and 50-304, dated March 8, 1985. Response to NRC requested for additional information dated February 19, 1985.

7. Letter from E. P. Rahe, Westinghouse, to D. G. Eisenhut, NRC, Westinghouse Letter NS-NRC-85-3025, dated March 22, 1985.

8. Letter from P. C. LeBlond, Nuclear Licensing Administrator, Commonwealth Edison, to H. R. Denton, Director, Nuclear Regulatory Commission, Subject: Zion Nuclear Power Station Units 1 and 2 Addendum to the Revised ECCS Analysis and Proposed Amendment to Facility Operating License Nos. DPR-39 and DPR-48 NRC Docket Nos. 50-295 and 50-304, dated April 19, 1985.

9. Letter from E. P. Rahe, Westinghouse, to C. Thomas, NRC, Westinghouse Letter NS-EPR-2952, dated August 14, 1984.