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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 51 TO FACILITY OPERATING LICENSE NO. DPR-30

COMMONWEALTH EDISON COMPANY

AND

IOWA-ILLINOIS GAS AND ELECTRIC COMPANY

QUAD CITIES STATION, UNIT NO. 2

DOCKET NO. 50-265

Introduction

By letter dated August 30, 1979 (Reference 1), and supplemented by Reference 2 and Reference 9, Commonwealth Edison (CE), the licensee, proposed amendments to Quad Cities Station Unit 2 (QC-2) License and Appendix A, Technical Specifications. CE has proposed these amendments to support its review of future reloads for QC-2 under the provisions of 10 CFR 50.59.

Our approval is only for the proposed amendment and does not constitute approval of CE's future reloads under the provisions of 10 CFR 50.59.

Evaluation

Safety Limit Critical Power Ratio (SLMCPR)

This change provides SLMCPRs in the Technical Specifications for all currently approved core loadings. With retrofit 8x8 fuel in the core the SLMCPR limit is specified as 1.07. Without retrofit 8x8 fuel the SLMCPR limit is 1.06. These limits have previously been found acceptable for this use in Reference 4 and on this basis the proposed change is acceptable.

Rod Drop Accident (RDA) Design Limit

The RDA design limit has been modified from 1.3%Δ maximum rod worth to 280 cal/gm peak fuel enthalpy rise. The 280 cal/gm design limit is acceptable per Standard Review Plan NUREG-75/087. Also, the power level below which the rod worth minimizer is required was increased from 10% to 20% of rated. This is conservative by comparison to the previous specification and is consistent with reactor safety analyses.

Maximum Average Planar Linear Heat Generation Rate (MAPLHGR)

New MAPLHGR curves from Reference 6 have been proposed. This employed methodology accepted for QC-2 in Reference 7. The only change to Reference 6 from that previously accepted is to incorporate prepressurized fuel in the analyses. This addition was performed in compliance with our guidelines in our acceptance of prepressurized fuel application (Reference 8). Therefore, the use of Reference 6 continues to be acceptable for QC-2 and the addition of Reference 6 MAPLHGR curves to the Technical Specification is also acceptable.

Linear Heat Generation Rate (LHGR) Power Spiking Penalty

The LHGR power spiking penalty for 8x8 fuel has been incorporated in the safety analyses by the reduction of the LHGR limit an equivalent amount. This change has been generically accepted by our Reference 5 letter.

Power Peaking

The next proposed change is to go from a total peaking factor formulation to a ratio of the fraction of limiting power density to fraction of rated power formulation for the local power peaking adjustment of neutron flux reactor protective system logic. This change in formulation has previously been approved for other BWRs, e.g.,Reference 3. These two formulations are identical in their results but the proposed formulation eliminates the need for different limits for different fuel types. Also the proposed change provides for increasing the Average Power Range Monitor (APRM) gains instead of a reduction in setpoints. This results in the same protective function.

Safety/Relief Valve (SRV) Setpoint

A reduction in the SRV safety function setpoint has been proposed. The SRV relief function setpoint was reduced to preclude multiple relief valves discharges. The proposed change would reduce the SRV safety function setpoint to the same value for consistency. The licensee plans to use this setpoint in future analyses. This reduction is conservative with respect to reactor vessel pressure relief function. It is also a small reduction ≈15 psi, so that there is no significant increase in valve demand, and, thus, probability of failure.

Water Level Setpoints

The current Technical Specifications have water level setpoints referenced to the top of the active fuel. Different active fuel lengths, as is the case for the 8x8 and 8x8R fuels, may confuse the specification and surveillance requirements. Therefore, the licensee has proposed to define the top of the active fuel as 360 5/16" above reactor vessel zero and the reactor low water level scram and ECCS initiation setpoints at 143 7/8" and 83 7/8" above the top of the active fuel, respectively. These definitions and setpoints are conservative values compared to those used in the reactor safety analyses.

These findings are based on current acceptable fuel assembly designs and any application of other designs would require specific justification of water level setpoints.

The reactor low water level scram setting had been established as 144 inches above the top of the core. With the licensee's agreement, we have modified the trip setpoint to be greater than 144 inches to assure that the scram occurs at or before the value assumed in reactor safety analyses.

Overpressure Protection Margin to Safety Valve Setpoints

The first proposed change is to delete the portion of the license restriction that requires reactor power level restrictions to maintain pressure margin to safety valve (SV) set points during the worst case pressurization transient. This restriction was imposed by the licensee to avoid an extensive outage in the event of SV discharge to the drywell. Our criteria for overpressurization protection (Standard Review Plan 5.2.2, NUREG-79/087) has been that "for the design basis normal operational transients, relief valve capacity must be sufficient to limit the pressure so as to prevent SV discharge directly to the containment," and, "for the most severe abnormal operational transient, with reactor scram, the SV capacity should be sufficient to limit the pressure to less than 110% of the reactor coolant pressure boundary design pressure." These criteria are satisfied by the proposed change.

Further, we do not consider the SV discharge to the drywell a safety concern, since all safety systems are to be qualified for LOCA environment which is more severe than the possible SV discharge. We have also reviewed BWR pressure relief systems operating experience (NUREG-0462) and have found that operating experience with SVs has been essentially failure free.

Coastdown Feedwater Heater Restrictions

In response to our concern on the potential for power operation outside the bounds of the transient analyses for coastdown operation, the licensee has proposed that the license restriction include a requirement to perform a safety evaluation if off-normal feedwater heater operation is needed (Reference 2). This requirement satisfies our concern.

Typographical Corrections and Clarification of Bases

The remaining changes fall into the category of typographical corrections and clarification of bases and do not, as such, represent a significant safety concern.

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 Section 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 the 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 consideration, (2) there is reasonable 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 the amendment will not be inimical to the common defense and security or to the health and safety of the public.

Dated: March 20, 1980

References

1. Letter from Cordell Reed (CECo), to the Director of Nuclear Reactor Regulation (USNRC) dated August 30, 1979.

2. Letter from D. Louis Peoples (CECo), to Director of Nuclear Reactor Regulation (USNRC) dated October 24, 1979.

3. Letter from T. A. Ippolito (USNRC) to G. T. Berry (Power Authority of the State of New York), dated November 22, 1978.

4. Letter from D. G. Eisenhut (USNRC) to R. Gridley (General Electric, GE) dated May 12, 1978.

5. Letter from D. G. Eisenhut (USNRC) to R. Gridley (GE) dated June 9, 1978.

6. "Loss-of-Coolant Accident Analysis Report for Dresden Units 2,3 and Quad Cities Units 1, 2 Nuclear Power Stations," NEDO-24164 A, dated April 1979.

7. Memorandum from R. L. Baer to D. L. Ziemann "Evaluation of Quad Cities Unit 2 for Cycle 4 Operation," dated February 21, 1978.

8. Letter from T. A. Ippolito (USNRC) to R. Gridley (GE), dated April 16, 1979.

9. Letter from D. L. Peoples (CECo) to Director, NRR, dated March 7, 1980.