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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 71 TO FACILITY OPERATING LICENSE NO. NPF-11 AND

AMENDMENT NO. 55 TO FACILITY OPERATING LICENSE NO. NPF-18

COMMONWEALTH EDISON COMPANY

LASALLE COUNTY STATION, UNITS 1 AND 2

DOCKET NOS. 50-373 AND 50-374

1.0 INTRODUCTION

In response to Generic Letter 88-11, "NRC Position on Radiation Embrittlement of Reactor Vessel Materials and its Effect on Plant Operations," the Commonwealth Edison Company (the licensee) requested permission to revise the pressure/temperature (P/T) limits in the LaSalle County Station Technical Specifications, Section 3.4. The request was documented in a letter from the licensee dated June 21, 1989. This revision also changes the effectiveness of the P/T limits of 16 and 32 effective full power years (EFPY). The licensee proposed to use one set of P/T limits for each unit. The proposed P/T limits were developed using Section 1 of Regulatory Guide (RG) 1.99, Revision 2. The proposed revision provides up-to-date P/T limits for the operation of the reactor coolant system during heatup, cooldown, criticality, and hydrotest.

To evaluate the P/T limits, the staff uses the following NRC regulations and guidance: Appendices G and H of 10 CFR Part 50; the ASTM Standards and the ASME Code, which are referenced in Appendices G and H; 10 CFR 50.36(c)(2); RG 1.99, Revision 2; Standard Review Plan (SRP) Section 5.3.2; and Generic Letter 88-11.

Each licensee authorized to operate a nuclear power reactor is required by 10 CFR 50.36 to provide Technical Specifications for the operation of the plant. In particular, 10 CFR 50.36(c)(2) requires that limiting conditions of operation be included in the Technical Specifications. The P/T limits are among the limiting conditions of operation in the Technical Specifications for all commercial nuclear plants in the U.S. Appendices G and H of 10 CFR Part 50 describe specific requirements for fracture toughness and reactor vessel material surveillance that must be considered in setting P/T limits. An acceptable method for constructing the P/T limits is described in SRP Section 5.3.2.

Appendix G of 10 CFR Part 50 specifies fracture toughness and testing requirements for reactor vessel materials in accordance with the ASME Code and, in particular, that the beltline materials in the surveillance capsules be tested in accordance with Appendix H of 10 CFR Part 50. Appandix H, in turn, refers to ASTM Standards. These tests define the extent of vessel embrittlement at the time of capsule withdrawal in terms of the increase in reference temperature. Appendix G also requires the licensee to predict the effects of neutron irradiation on vessel embrittlement by calculating the adjusted reference temperature (ART) and Charpy upper shelf energy (USE). Generic Letter 88-11 requested that licensees and permittees use the methods in RG 1.99, Revision 2, to predict the effect of neutron irradiation on reactor vessel materials. This guide defines the ART as the sum of unirradiated reference temperature, the increase in reference temperature resulting from neutron irradiation, and a margin to account for uncertainties in the prediction method.

Appendix H of 10 CFR Part 50 requires the licensee to establish a surveillance program to periodically withdraw surveillance capsules from the reactor vessel. Appendix H refers to the ASTM Standards which, in turn, require that the capsules be installed in the vessel before startup and that they contain test specimens made from plate, weld, and heat-affected-zone (HAZ) materials of the reactor beltline.

2.0 EVALUATION

The staff evaluated the effect of neutron irradiation embrittlement on each beltline material in the LaSalle 1 and 2 reactor vessels. The amount of irradiation embrittlement was calculated in accordance with Section 1 of RG 1.99, Revision 2. The staff has determined that the material with the highest ART at 16 EFPY for LaSalle 1 was lower shell plate C5978-2 with 0.11% copper (Cu), 0.59% nickel (Ni), and an initial RT_{NDT} of 23°F. The material with the highest ART at 16 EFPY for LaSalle 2 was lower-intermediate shell plate C9404-2 with 0.07% Cu, 0.49% Ni, and an initial RT_{NDT} of 52°F.

The licensee has not removed any surveillance capsules from LaSalle 1 or 2. The staff has checked the contents of all surveillance capsules and has ascertained that they contain Charpy impact specimens and tensile specimens made from base metal, weld metal, and HAZ metal.

For the limiting beltline material in LaSalle 1, plate C5978-2, the staff calculated the ART to be 42.3°F at 1/4T (T = reactor vessel beltline thickness) for 16 EFPY, and at 3/4T, 33.6°F. The staff used a neutron fluence of 1.3E17 n/cm^2 at 1/4T and 5.0E16 n/cm^2 at 3/4T. For the limiting beltline material in LaSalle 2, plate C9404-2, the staff calculated the ART at 16 EFPY to be 64.3°F at 1/4T for a neutron fluence of 1.4E17 n/cm^2 and 59.4°F at 3/4T for a neutron fluence of 6E16 n/cm^2.

For 32 EFPY, the limiting material in LaSalle 1 are vertical welds 3-308 A, B, and C with 0.37% Cu, 0.75% Ni, and an initial RT_{NDT} of -30°F. The staff calculated the ART at 1/4T to be 66.0°F and 25.7°F at 3/4T. The neutron fluences used were 2.6E17 and 1.0E17 n/cm^2, respectively. For LaSalle 2, the limiting material at 32 EFPY is still the lower-intermediate shell plate C9404-2 with an ART at 1/4T of 70.9° and at 3/4T of 63.9°F. The neutron fluences used were 2.8E17 and 1.2E17 n/cm^2, respectively.

The licensee used the method in Section 1 of RG 1.99, Revision 2, to calculate ART values for the inner diameter (I.D.) at 16 and 32 EFPY for LaSalle 1 and 2. The licensee's values agree with the staff's values for ART at the I.P. at 16 and 32 EFPY for both units. Substituting the ART values described above into equations in SRP 5.3.2, the staff verified that the proposed P/T limits for heatup, cooldown, and hydrotest meet the beltline material requirements in Appendix G of 10 CFR Part 50.

In addition to beltline materials, Appendix G of 10 CFR Part 50 also imposes P/T limits based on the reference temperature for the reactor vessel closure flange materials. Section IV.2 of Appendix G states that when the pressure exceeds 20% of the preservice system hydrostatic test pressure, the temperature of the closure flange regions highly stressed by the bolt preload must exceed the reference temperature of the material in those regions by at least 120°F for normal operation and by 90°F for hydrostatic pressure tests and leak tests. Based on the flange reference temperature of 40°F, the staff has determined that the proposed P/T limits satisfy Section IV.2 of Appendix G.

Section IV.B of Appendix G requires that the predicted Charpy USE at end of life be above 50 ft-lb. At the time the LaSalle 1 and 2 reactor vessels were ordered, there were no requirements that the USE data exist. In SSER No. 2 (Reference 8), the staff reported that the licensee supplied data and analyses to demonstrate that all the beltline materials in LaSalle 1 attain an unirradiated USE of greater than 75 ft-lb. Using an EOL fluence of 3.9E17 n/cm^2 with Figure 2 of RG 1.99, Revision 2, the staff has calculated that a weld with an unirradiated USE of 75 ft-lb would have a USE of 56 ft-lb after irradiation. This is greater than 50 ft-lb and, therefore, is acceptable.

For LaSalle 2, to assure that the 50 ft-lb requirement of Section IV.B of Appendix G is met, the staff determined that the unirradiated USE required was 57 ft-lb for welds and 58 ft-lb for plate materials. In SSER No. 2 (Ref. 8), the staff reported that the licensee supplied data and analyses to demonstrate that both required unirradiated USE values have been met for all the beltline materials.

Further, it was requested that the 200°F limit for COLD SHUTDOWN be raised to 212°F for the purpose of performing hydrostatic or leak testing or heat-up by non-nuclear means. Raising this limit to 212°F is acceptable because the LaSalle Station is only slightly above sea level so the boiling point of water is approximately 212°F which assures that the reactor coolant, under only slight pressure, will not boil during these tests. While possibly creating some technical difficulty with the testing program, boiling under these circumstances has no safety significance because all the control rods will be fully inserted, thus creating no potential for criticality.

3.0 ENVIRONMENTAL CONSIDERATION

These amendments involve changes to requirements with respect to installation or use of a facility component located within the restricted area, as defined in Part 20. The staff has determined that this amendment involves 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 meet the eligibility criteria for categorical exclusion set forth in 10 CFR 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 this amendment.

4.0 CONCLUSION

The staff concludes that the proposed P/T limits for the reactor coolant system for heatup, cooldown, leak test, and criticality are valid through 16 and through 32 EFPY for both LaSalle 1 and 2 because the limits conform to the requirements of Appendices G and H of 10 CFR Part 50. The licensee's submittal also satisfies Generic Letter 88-11 because the licensee used the method in RG 1.99, Revision 2 to calculate the ART. Hence, the proposed P/T limits may be incorporated into the LaSalle 1 and 2 Technical Specifications including the proposed modifications to the Bases.

The staff has 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.

5.0 REFERENCES

1. Regulatory Guide 1.99, Radiation Embrittlement of Reactor Vessel Materials, Revision 2, May 1988.

2. NUREG-0800, Standard Review Plan, Section 5.3.2, Pressure-Temperature Limits.

3. LaSalle Safety Evaluation Report Plan, Supplements 1 and 2.

4 T.A. Caine, "LaSalle County Station Units 1 and 2 Fracture Toughness Analysis per 10 CFR 50 Appendix G, SASR 88-10," General Electric, March 1988.

5. June 21, 1989, Letter from W. E. Morgan (CE) to T.E. Murley (USNRC), Subject: LaSalle County Station--Units 1 and 2, Application for Amendment to Facility Operating Licenses NPF-11 and NPF-18, Revision of Pressure-Temperature Curves.

6. LaSalle Final Safety Analysis Report, Section 5.2 and Technical Specifications.

7. NUREG-0519, Safety Evaluation Report Related to the Operation of LaSalle County Station Units 1 and 2, March 1981.

8. NUREG-0519, Supplement 2, Safety Evaluation Report Related to the Operation of LaSalle County Station Units 1 and 2, February 1982.

Principal Contributor: John C. Tsao NRR/EMTB

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