UNITED STATES
NUCLEAR REGULATORY COMMISSION
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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION RELATED TO RADIATION EMBRITTLEMENT OF REACTOR VESSEL MATERIALS AND ITS IMPACT ON PLANT OPERATIONS

HOUSTON LIGHTING & POWER COMPANY

DOCKET NOS. 50-498 AND 50-499

SOUTH TEXAS PROJECT, UNITS 1 AND 2

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 Houston Lighting and Power Company (the licensee) stated that the STP plant specific pressure-temperature (P/T) curves currently in use are more conservative than the requirements of Regulatory Guide (RG) 1.99, Revision 2. Hence no further action was planned. The response was documented in a letter from the licensee dated November 22, 1989 (ST-HL-AE-2858). The licensee proposed no change to the effectiveness of the P/T limits of 32 effective full power years (EFPY). In addition, the licensee proposed to use the more conservative set of P/T limits for the operation of the reactor coolant system during heatup, cooldown, criticality, and hydrotest of both units.

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, Rev. 2; Standard Review Plant (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. Appendix 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 re-

quires 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, Rev. 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 South Texas 1 and 2 reactor vessels. The amount of irradiation embrittlement was calculated in accordance with RG 1.99, Rev. 2. The staff has determined that the material with the highest ART at 32 EFPY for both units was the intermediate shell plate (R1606-3) in Unit 1 with 0.05% copper (Cu), 0.62% nickel (Ni), and an initial RT_{ndt} of 10°F.

The licensee has not removed any surveillance capsules from South Texas 1 and 2. According to the licensee's Final Safety Analysis Report (FSAR), all surveillance capsules contained Charpy impact specimens and tensile specimens made from base metal, weld metal, and HAZ metal.

For the limiting beltline material, intermediate shell plate R1606-3, the staff calculated the ART to be 82.3°F at 1/4T (T = reactor vessel beltline thickness) for 32 EFPY. The staff used a neutron fluence of 3.96E19 n/cm^2 at 1/4T. The ART was determined using Section 1 of RG 1.99, Rev. 2.

The licensee used the method in RG 1.99, Rev. 2, to calculate an ART of 80°F at 32 EFPY at 1/4T for the same limiting intermediate shell plate material. The staff judges that a difference of 2.3°F between the licensee's ART of 80°F and the staff's ART of 82.3°F is acceptable. Substituting the ART of 82.3°F into equations in SRP 5.3.2, the staff verified that the present 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 $0^{o}F$, the staff has determined that the present 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. Based on data from the licensee's FSAR, the lowest measured Charpy USE is 94 ft-lb for intermediate shell plate R1606-2 in Unit 1. Using the method in RG 1.99, Rev. 2, the predicted Charpy USE of the plate material at the end of life will be 69.6 ft-lb. This is above 50 ft-lb and therefore is acceptable.

3.0 CONCLUSION

The staff concludes that the present P/T limits for the reactor coolant system for heatup, cooldown, leak test, and criticality are valid through 32 EFPY 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, Rev. 2 to calculate the ART. Hence, the present P/T limits in the South Texas 1 and 2 Technical Specifications may be retained.

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Principal Contributor: John Tsao