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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO FEEDWATER NOZZLE CRACKING

GEORGIA POWER COMPANY, ET AL.

HATCH NUCLEAR PLANT, UNIT 2

DOCKET NO. 50-366

1.0 INTRODUCTION

The Georgia Power Company's (GPC or licensee) letter, dated September 30, 1991, enclosed the General Electric Company's (GE) report, GE-NE-523-95-0991 (Reference 1), which contained the final feedwater nozzle fracture mechanics assessment for Hatch Unit 2. The intent of this report was to demonstrate full compliance with NRC recommendations regarding feedwater nozzle crack growth, as specified in NUREG-0619 (Reference 2), and amended by Generic Letter 81-11 (Reference 3).

From 1974 to 1980, inspection of the feedwater nozzle/sparger systems disclosed some degree of cracking in the bore and inner radii of the reactor vessel feedwater nozzles in 18 of the 23 commercially operated boiling water reactor (BWR) plants in the United States. The NRC staff has reviewed this issue as part of Generic Technical Activity A-10, and its recommendations are contained in NUREG-0619.

In the above NUREG, the staff concluded that crack initiation in feedwater nozzles was caused by high-cycle thermal fatigue. From analyses and experience in repairing feedwater nozzles, it is generally known that high-cycle thermal fatigue cracks propagate to a depth of about 0.25 inch before the cyclic thermal stress amplitude attenuates to an insignificant level. Analyses also indicate that stainless steel cladding contributes to the high-cycle thermal fatigue crack initiation. Furthermore, the staff concluded that significant propagation of the small high-cycle thermal fatigue cracks would result from low-frequency and high-amplitude stresses, which are caused by the intermittent flow of cold feedwater into the vessel during startup and shutdown, and during hot standby conditions, when cold feedwater is added to maintain reactor water level. The frequency and magnitude of the stresses depend, to a large degree, on whether such additions are modulated or are made by an on-off flow control system.

The staff's position in NUREG-0619 was that improvements should include: (1) nozzle clad removal, (2) installation of improved sparger design, and (3) system changes. The system changes include a low-flow controller having the six characteristics described in Section 3.4.4.3 of the GE Report NEDE-21821-A (Reference 4), and rerouting of the reactor water cleanup (RWCU) system to all feedwater nozzles. As a result of comments received from GE and others, the staff, in Generic Letter 81-11, clarified its position relative to the installation of low-flow controllers. The staff indicated in this generic letter that continued use of existing controllers is acceptable, provided a plant-specific fracture mechanics analysis or application of the analysis, already existing in Section 4 of NEDE-21821-A, does not result in the growth of a crack to greater than one inch during the 40 year life of the plant.

GPC stated that the Hatch Unit 2 feedwater nozzles were unclad, the spargers were welded-in, and the RWCU system was rerouted to all feedwater nozzles. Therefore, the only unresolved item for Hatch Unit 2 was related to the flow controller modifications and the fracture mechanics analysis which demonstrates that the feedwater nozzle crack growth is less than one inch in 40 years.

To resolve this item, you provided the following: (1) a fracture mechanics analysis in the GE report, NEDC-30256 (Reference 5), (2) a response to NRC staff's questions in SASR 86-38 (Reference 6), and (3) an updated fracture mechanics analysis report, SASR 1290-HT2 (Reference 7). The dates of NRC correspondence regarding the above submittals are: January 21 and December 8, 1986, and June 28, 1991, respectively.

The staff has reviewed the fracture mechanics analysis contained in the GE report, GE-NE-523-95-0991, and its safety evaluation is provided below.

2.0 EVALUATION

According to NUREG-0619, cracks in the feedwater nozzle are caused by high-cycle thermal stresses in the crack initiation phase, and low-frequency high-amplitude stresses in the crack propagation phase. These low-frequency high-amplitude stresses are caused by intermittent flow of cold feedwater into the vessel during startup, shutdown, and hot standby conditions.

The GE fracture mechanics fatigue analysis assumes the existence of a 0.25 inch deep flaw at the feedwater location where the peak combined pressure and thermal stresses are expected to occur. Because stresses from the high-cycle thermal fatigue attenuate to an insignificant level at 0.25 inch, an assumed initial flaw of 0.25 inch deep would conservatively envelop the initial crack size.

The amount of crack growth from the low-cycle thermal fatigue stresses was evaluated for a 40 year period. The thermal events and cycles within this period were constructed as follows: (1) actual plant data were used for the 1979-1982, and the 1985-1991 time periods, (2) the thermal events and cycles for the 1979-1982 period were extrapolated to cover the 1982-1985 period where adequate data were not available, and (3) the 1985-1991 events were used to predict cycle counts for the remainder of the 40 year plant life. During the 1985-1991 period, the licensee had recorded the number of thermal cycles, feedwater flow and temperature for startups, shutdowns and scrams to the hot standby conditions. The staff finds that this approach of constructing the plant operating history is reasonable and acceptable. The staff also agrees that neglecting temperature and pressure fluctuations less than 25°F and 100 psi in defining the thermal and pressure cycles for each event, should not have a significant impact on the crack growth evaluation.

Finite element computer codes were used to develop thermal and pressure stresses in the feedwater nozzle during startup, shutdown, and scram conditions. Heat transfer coefficients and annulus boundary temperatures in the nozzle were developed from the Moss Landing Test Data pertaining to the feedwater nozzle configuration of Hatch Unit 2, as documented in NEDC-30256. GE stated that these values remain unchanged and, thus, were used again in the current analysis. These heat transfer coefficients and boundary temperatures were accepted by the staff in its safety evaluation dated December 8, 1986, and are, again, considered acceptable.

The thermal and pressure stresses were converted to stress intensity factors using the methods reported in Section 4 of NEDE-21821-A. The stress intensity factors were used to predict the amount of crack growth. The amount of crack growth per cycle was calculated using the fatigue crack growth data for low alloy steels from Section XI of the ASME Code. The amount of fatigue crack growth is dependent upon the changes in stress intensity factors resulting from the change in pressure and thermal stresses. The analysis performed by the licensee and GE has considered the effect of pressure and thermal stresses on the amount of crack growth at the nozzle location with the highest combined stress. The staff has reviewed the empirical formula used for stress intensity factor calculations and finds that the formula is acceptable because it had been validated against analytical results from a three dimensional finite element nozzle model.

The projected final crack length at the end of the plant's life was calculated by first evaluating the incremental crack growth for each cycle. The crack size was then updated and the procedure was repeated for all cycles until all events had been analyzed. Since credible computer output is expected once the input of cycles and events is defined correctly, the staff finds that the final crack length of 0.96 inch is acceptable.

3.0 CONCLUSION

Based upon the number of events, cycles and thermal conditions projected to occur in the remaining life (i.e., about 26 years) of Hatch Unit 2, and using Section XI of the ASME Code, the amount of final crack length in the feedwater nozzles at the end of Hatch Unit 2 life (i.e., 40 years) was predicted by the licensee and GE to be 0.96 inch. Thus, the staff concludes that the crack growth analysis for the feedwater nozzle is in compliance with NUREG-0619, as amended by Generic Letter 81-11.

4.0 REFERENCES

1. GE-NE-523-95-0991, "Updated Feedwater Nozzle Fracture Mechanics Analysis for Edwin I. Hatch Nuclear Power Station, Unit 2," General Electric Company, September 1991, transmitted by GPC's letter dated September 30, 1991.

2. NUREG-0619, "BWR Feedwater Nozzle and Control Rod Drive Return Line Nozzle Cracking," U.S. Nuclear Regulatory Commission, November 1980.

3. Generic Letter 81-11 to all Power Reactor Licensees from Darrell Eisenhut, U.S. Nuclear Regulatory Commission, February 28, 1981.

4. NEDE-21821-A, "Boiling Water Reactor Feedwater Nozzle/Sparger Final Report," General Electric Company, February 1980.

5. NEDC-30256, "Edwin I. Hatch Nuclear Power Station, Unit 2, Feedwater Nozzle Fracture Mechanics Analysis," General Electric Company, August 1983, transmitted by GPC's letter dated September 19, 1984.

6. SASR 86-38, "Response to the NRC Questions with Regard to the Hatch 2 Feedwater Nozzle NUREG-0619 Report," General Electric Company, June 16, 1986, transmitted by GPC's letter dated July 28, 1986.

7. SASR 1290-HT2, "Updated Feedwater Nozzle Fracture Mechanics Analysis for Edwin I. Hatch Nuclear Power Station, Unit 2," General Electric Company, December 1990, transmitted by GPC's letter dated January 21, 1991.

8. ASME Boiler and Pressure Vessel Code, Section XI, "Rules for In-Service Inspection of Nuclear Power Plant Components," Appendix A, "Analysis of Flaws," 1989 Edition.

Principal Contributor: Simon C. F. Sheng, EMCB, NRR

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