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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 120

TO PROVISIONAL OPERATING LICENSE NO. DPR-16

GPU NUCLEAR CORPORATION AND
JERSEY CENTRAL POWER & LIGHT COMPANY

OYSTER CREEK NUCLEAR GENERATING STATION

DOCKET NO. 50-219

INTRODUCTION

By letter dated January 19, 1988, the GPU Nuclear Corporation (the licensee) proposed to revise the pressure-temperature limits in the Oyster Creek Nuclear Generating Station Technical Specifications through 15 effective-full-power years (EFPY). The proposed pressure-temperature limits were developed from the licensee's submittal, "Testing and Evaluation of Irradiated Reactor Vessel Materials Surveillance Program Specimens," TDR-725. The limits consist of three curves that set minimum pressure and temperature for three operating conditions - hydrostatic and leakage test, heatup or cooldown (core not critical), and heatup or cooldown (core critical). Presently, the plant is about to reach 10 EFPY which is the current technical specification limit for the pressure-temperature curves. The proposed new curves will allow the operator to operate the reactor continuously through 15 EFPY without violating the Technical Specifications.

DISCUSSION

Part of the NRC's effort to ensure integrity of the reactor vessel is to periodically evaluate the reduction in fracture toughness of the vessel material due to neutron irradiation embrittlement. The effort consists of three steps.

First, the licensee is required to establish a surveillance program in accordance with Appendix H of 10 CFR 50, which requires periodic withdrawal of surveillance capsules from the reactor vessel. The capsules are installed in the vessel prior to startup and they should contain test specimens that were made from the plate, weld, and heat affected zone materials of the reactor beltline.

Secondly, the licensee is required to perform Charpy impact tests, tensile tests, and neutron fluence measurements of the specimens. These tests define the condition of vessel embrittlement at the time of capsule withdrawal in terms of the shift of the reference temperature, RT_{NDT}, and upper shelf energy. The licensee should also predict the future vessel embrittlement by calculating the adjusted RT_{NDT} and upper shelf energy at a specific EFPY. The licensee may use either Revision 1 or draft Revision 2 of Regulatory Guide 1.99 to calculate the adjusted RT_{NDT}. The upper shelf energy is the average energy value for all specimens whose test temperature is above the upper end of the transition temperature region. The licensee is required by 10 CFR 50 Appendix G to assure that the adjusted RT_{NDT} will not exceed 200°F and that the upper shelf energy will not be below 50 ft-lb at the end of plant life.

Thirdly, the licensee is required to develop a set of pressure-temperature curves based on the adjusted RT_{NDT} of the limiting vessel material. The curves should satisfy the recommended methods and requirements described in 10 CFR 50, Appendix G and Standard Review Plan 5.3.2.

EVALUATION

The Oyster Creek Nuclear Station is a boiling water reactor which has an inside diameter of 213 inches and mean wall thickness of 7.125 inches. The reactor vessel was fabricated from ASTM A302, Grade B plate material. The submerged arc weld materials were RACO #3 bare wire and ARCO B-5 flux. Manual metal arc welding used 8018 covered electrodes.

General Electric installed three specimen capsules as a part of the reactor vessel surveillance program. The withdrawal of the first capsule in 1971 was unsuccessful. Capsule No. 2 was withdrawn in March 1984 at 8.38 EFPY. After examining specimens in capsule No. 2, the licensee found several material discrepancies and that the program does not meet requirements of 10 CFR 50 Appendix H. For example, the limiting material and the beltline welds were not included in the capsule. The exact copper and nickel contents of several plates and welds were unavailable. These discrepancies were partly due to the vintage of the plant and partly because the surveillance program was initiated before the issuance of Appendix H. Nevertheless, the staff had reviewed the surveillance program under the Systematic Evaluation Program guidelines in the early 1980's and found it acceptable. In this evaluation, the staff concentrated not on review of the program itself but on the pressure-temperature curves.

The specimen capsule data showed that the G-308-1 plate had a RT_{NDT} shift of 72°F measured at 30 ft-lb transition temperature and had received a neutron fluence of 7.46×10^{17} n/cm^2. Since the G-308-1 plate showed a higher RT_{NDT} shift than that of the weld and heat-affected-zone materials in the capsule, the data of the G-308-1 plate were used in the RT_{NDT} calculation of the limiting material.

The licensee used Regulatory Guide 1.99, draft Rev. 2 to calculate the adjusted RT_{NDT} because the Rev. 1. calculation showed a lower RT_{NDT} shift.

To calculate the highest adjusted RT_{NDT}, the licensee compared the copper and nickel contents of the G-308-1 plate to those unirradiated specimens of five other beltline plates not placed in the capsule. The licensee conservatively applied the chemistry factor, neutron fluence and measured RT_{NDT} of the G-308-1 plate to one of the five plate specimens that had the worst combination of copper content, nickel content, and initial RT_{NDT}. The calculation showed that the G-8-6 plate had the highest adjusted RT_{NDT} of 125°F at the neutron fluence of 1.11×10^{18} n/cm^2, 15EFPY, and 1/4T (vessel thickness) location. The G-8-6 plate was selected as the limiting material.

The licensee also predicted the end-of-life adjusted RT_{NDT} of 142°F and the upper shelf energy of 61.5 ft-lb at a neutron fluence of 2.38×10^{18} n/cm^2 for the G-8-6 plate. These values satisfy the 10 CFR 50 Appendix G requirements.

To construct the pressure-temperature curves, the licensee followed closely the method described in NRC's Standard Review Plan 5.3.2 and ASME Section III Appendix G except in the membrane stress calculation. To calculate the membrane stress, the licensee used the "vessel radius-thickness" relationship whereas SRP 5.3.2 prescribed the "allowable stress-design pressure" relationship. The former gives a lower temperature profile than that of the latter; but, the former method is not necessarily incorrect. The staff determined that the licensee's method was acceptable based on the stress analysis of a cylindrical container having a large radius-to-thickness ratio. (Ref. Roark, R.J., "Formulas for Stress and Strain," 4th edition, page 308). The lower part of the pressure-temperature curves also has to satisfy the specific requirements of 10 CFR 50 Appendix G for boiling water reactors because the boiling water reactor vessel has an inherent pressure-temperature limitation when the reactor water level is within the normal range for power operation and the reactor pressure is less than 10 percent of the preservice system hydrostatic test pressure. The pressure-temperature curve is limited by the closure flange regions that are highly stressed by the bolt preload. The minimum permissible temperature should be 60°F above the initial RT_{NDT} of the flange and when the test pressure is above 20% of the hydrotest pressure, the permissible temperature should be 90°F above the initial RT_{NDT}. Based on an initial RT_{NDT} of 40°F for the Oyster Creek reactor flange, the minimum temperature should be 100°F and the permissible test temperature should be 130°F. Examining the lower part of the pressure-temperature curves, the staff determines that the curves satisfy the 10 CFR 50 Appendix G requirements.

The staff has reviewed the proposed pressure-temperature curves and corresponding paragraphs in the Technical Specifications. The licensee has applied appropriately Regulatory Guide 1.99, draft Rev. 2, 10 CFR 50 Appendix G, and Standard Review Plan 5.3.2 to calculate the adjusted RT_{NDT} and to develop the pressure-temperature curves. The staff concludes that the proposed pressure-temperature curves are valid through 15 EFPY and may be incorporated into the Oyster Creek Nuclear Station Technical Specifications.

ENVIRONMENTAL CONSIDERATION

This amendment changes a requirement with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The staff has determined that the 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 this amendment involves no significant hazards consideration and there has been no public comment on such finding. Accordingly, the amendment meets 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.

CONCLUSION

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 the amendment will not be inimical to the common defense and security nor to the health and safety of the public.

Dated: March 21, 1988

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