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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 15 TO FACILITY OPERATING LICENSE NO. DPR-18

ROCHESTER GAS AND ELECTRIC CORPORATION

R. E. GINNA NUCLEAR POWER PLANT

DOCKET NO. 50-244

INTRODUCTION

By letter dated December 8, 1982 and as supplemented October 10, 1983, August 8, 1984 and August 19, 1985, the Rochester Gas and Electric Corporation (RG&E or the licensee) submitted a proposed license amendment for Facility Operating License No. DPR-18 for the R. E. Ginna Nuclear Power Plant (the facility). The amendment changes the Technical Specifications (TS) to extend the reactor vessel pressure-temperature limits from 10.6 to 21.0 effective full power years (EFPY). The amendment would also permit the licensee to withdraw the next reactor vessel surveillance capsule at 17 EFPY; this based on the analysis of reactor vessel capsule "T" which was previously withdrawn.

DISCUSSION

This evaluation was conducted in two parts; first, the core physics aspects needed to support the licensee's dosimetry analyses of Capsule T, and secondly, the material fracture toughness aspects needed to verify neutron fluence to critical welds.

The fracture toughness analysis is then related to the proposed pressure-temperature limits, shown as heatup rate and cooldown rate curves in the TS. Therefore, the following evaluation is presented in two separate sections, namely: Dosimetry and Material Fracture Toughness.

EVALUATION OF DOSIMETRY

RG&E has submitted an application to amend the Ginna operating license based on the results of the analysis of surveillance capsule T. The analysis was performed by Westinghouse (W) and was published as WCAP-10086 (Ref. 1). Staff review of WCAP-10086 resulted in a request for additional information which was related to the power distribution, methodology uncertainty and the transport approximations used in the analysis (Ref. 2). These issues were addressed in letters to D. Crutchfield on October 10. 1983 and to W. Paulson on August 8, 1984 (Refs. 3 and 4).

The licensees letter of October 10, 1983 provided responses to the concerns expressed by the staff in Reference 2. A summary of the concerns and responses follows:

- Concern: The use of an average generic power distribution instead of a plant specific distribution should be justified.

  Response from licensee: A set of calculations has been performed with the plant specific actual core burnup information to generate plant specific fluence levels on a cycle-by-cycle basis.

- Concern: The updating of the results of previous capsules R and V should be discussed and justified.

  Response: Dosimetry data from capsules V, R and T based on plant specific analyses were shown to be in good agreement with the experimental data.

- Concern: The benchmarking of the discrete ordinates analyses procedures should be established.

  Response: The transport methodology has been benchmarked against the ORNL Pool Critical Assembly (PCA) facility results as well as against the Westinghouse power reactor surveillance capsule data.

- Concern: An analysis of the error and uncertainty bounds should be provided.

  Response: When plant specific power distributions are used, the benchmarking studies show that fluence predictions are within ±15% of the measured values of the surveillance capsule locations. These predictions tend to be in good agreement with the calculations based on the generic power distribution.

- Concern: The use of the P_1 approximation should be justified.

  Response: Neutron transport calculations in the R-Θ geometry were carried out using the DOT discrete ordinates code and the SAILOR cross section library. The SAILOR library is a 47 group ENDF/B-IV based set. Anisotropic neutron scattering is treated with a P_3 expansion of the cross sections. A flux calculation was performed using the P_3 cross section which provided a direct comparison to the P_1 results.[3] The results are acceptable; for example, for Cycle 14 with plant specific data, the P_3/P_1 azimuthal ratios range from -8% to +14%.

Further information was provided by the licensee in his August 8, 1984 letter to support the license amendment application of December 8, 1982. Staff review of this information submitted in support of the dosimetry analyses of capsule T for the Ginna plant has been reviewed and the data has been found satisfactory and acceptable. We conclude that the dosimetry analysis is acceptable.

References to Dosimetry Evaluation

1. S. E. Yanichko, et al., "Analysis of Capsule T from the Rochester Gas and Electric Corporation, R. E. Ginna Nuclear Power Plant Reactor Vessel Radiation Surveillance Program," WCAP-10086, April 1982.

2. Memorandum, W. Johnston to F. Miraglia, "Rochester Gas and Electric Corporation, R. E. Ginna Nuclear Plant, Unit 1" dated April 8, 1983.

3. Letter from J. Maier to D. Crutchfield, dated October 10, 1983.

4. Letter-report, R. Kober to W. Paulson, dated August 8, 1984.

EVALUATION OF MATERIAL FRACTURE TOUGHNESS

The Rochester Gas and Electric Corporation in a letter from J. E. Maier to H. R. Denton dated December 8, 1982 requested that the Technical Specifications for the R. E. Ginna Nuclear Power Plant, Unit No. 1 (hereafter Ginna) be revised to increase the effectivity of the reactor vessel pressure-temperature limits to 21.0 effective full power years (EFPY) and to permit withdrawal of the next reactor vessel surveillance capsule at 17 EFPY. The licensee indicates that the bases for the revised technical specification was the material test results from Capsule "T" of the R. E. Ginna Nuclear Plant Reactor Vessel Radiation Surveillance Program. The test results are reported in Westinghouse Report WCAP-10086.

The change in reactor vessel pressure-temperature limits depends upon the amount of neutron irradiation damage received by the limiting reactor vessel beltline material. The amount of neutron irradiation damage is estimated by performing Charpy V-notch (CVN) impact tests on unirradiated and irradiated material. The material property measured in this test is the adjusted reference temperature, ΔRT_{NDT}.

In the Ginna reactor vessel the limiting beltline material is the intermediate to lower shell weld which is identified as SA-847. The weld metal in the Ginna surveillance capsules is weld metal SA-1036, which was prepared using the same heat of wire (61782) and flux type (Linde 80) as weld metal SA-847, but not the same flux heat. Since SA-1036 and SA-847 weld metals were fabricated from the same heat of wire and flux type, the staff considers that the ΔRT_{NDT} for the SA-1036 weld metal will be representative of the ΔRT_{NDT} for SA-847 weld metal.

Table I compares the predicted ΔRT_{NDT} for SA-1036 weld metal using the Guthrie Mean and the Regulatory Guide 1.99, Rev. 1 formula to the ΔRT_{NDT} for SA-1036 weld metal from the Ginna Surveillance Capsules V, R and T. This comparison indicates that the Guthrie mean formula best predicts the ΔRT_{NDT} for the SA-1036 weld metal. The staff uses the Guthrie mean formula, which is reported in Commission Report SECY 82-465, and the Regulatory Guide 1.99, Rev. 1, to predict the ΔRT_{NDT}, because of the measurement variability in the CVN impact test and neutron dosimetry.

The Guthrie mean formula utilizes the amount of copper and nickel in the weld to predict the ΔRT_{NDT} as a function of neutron fluence. The amount of copper and nickel in a weld metal depends primarily on the weld wire chemical composition and the amount of copper plating on the weld wire. Since the amount of copper plating varies along a length of wire, the amount of this element in a weld must be estimated from a statistical study of the weld cross-section. This study has been performed by the Ginna reactor vessel fabricator, Babcock & Wilcox, on weld metal which was fabricated using the same heat of weld wire (heat 61782) as was used in the SA-847 weld metal. The results of this study is reported in B&W Proprietary Report BAW 1511P. The staff considers that the chemical composition for SA-847 weld metal is accurately described by the chemical composition for the welds fabricated from heat No. 61782 weld wire, which is reported in Report BAW 1511P.

The pressure-temperature limits have been evaluated using the method documented in Standard Review Plan 5.3.2. The ΔRT_{NDT} for the SA-847 weld metals was estimated using the Guthrie mean formula and the chemical composition for welds fabricated from heat 61782 weld wire, which is reported in B&W Report BAW 1511P. The results of our review indicates that the proposed pressure-temperature limits are acceptable until the intermediate to lower shell weld accumulates a neutron fluence of 1.5×10^{19} $^{n}/cm^{2}$ (E> 1MEV). Based on our evaluation of the licensee's data in his August 19, 1985 supplemental submittal we conclude that the critical weld will not accumulate a fluence of 1.5×10^{19} $^{n}/cm^{3}$ for 21 EFPY; therefore, the proposed heatup and cooldown rate curves are acceptable.

The licensee has requested that the date for removal of the next reactor vessel surveillance capsule be revised to the refueling outage which corresponds to 17 EFPY. According to WCAP 10086, 17 EFPY corresponds to a capsule neutron fluence of 4.10×10^{19} $^{n}/cm^{2}$, which is the approximate fluence at the inner surface location at the end-of-life of the reactor vessel. The removal of the next reactor vessel material surveillance capsule when its fluence reaches the value estimated for the inner surface location at the reactor vessel end-of-life is considered by the staff acceptable.

Table I

Comparison of Ginna Surveillance Capsules ΔRT_{NDT} to the ΔRT_{NDT} Predicted Using the Guthrie Mean Formula and the Regulatory Guide 1.99 Rev. 1 Formula

Capsule	Fluence ($^{n}/cm^{2} \times 10^{19}$)	ΔRT_{NDT}(°F) Capsule	ΔRT_{NDT}(°F) Guthrie Mean(1)	ΔRT_{NDT}(°F) Reg. Guide 1.99(1)
V	.703	140	130	176
R	1.01	165	143	211
T	1.75	150	166	278

(1)Weld Metal Chemistry of Capsule R utilized in calculation.

ENVIRONMENTAL CONSIDERATION

This amendment involves changes to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and a surveillance requirement. We have 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 radiation exposure. The NRC staff has made a proposed determination that the amendment involves no significant hazards consideration, and there has been no public comment on such finding. Accordingly, this 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 Commission made a proposed determination that this amendment involved no significant hazards consideration which was published in the Federal Register (48 FR 49595) on October 26, 1983 and consulted with the state of New York. No public comments were received, and the State of New York did not have any comments.

We have 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 this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributors:
L. Lois
B. Elliot
M. Fairtile

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