PDR

June 12, 1986

Docket No. 50-244

Mr. Roger W. Kober, Vice President
Electric and Steam Production
Rochester Gas & Electric Corp.
89 East Avenue
Rochester, New York 14649

Dear Mr. Kober:

The Commission has issued the enclosed Amendment No. 15 to Facility Operating License No. DPR-18 for the R. E. Ginna Nuclear Power Plant. This amendment is in response to your application dated December 8, 1982 and as supplemented October 10, 1983, August 8, 1984 and August 19, 1985.

The amendment changes the Technical Specifications to extend the reactor vessel pressure-temperature limits from 10.6 to 21.0 effective full power years (EFPY) and permits withdrawal of the next reactor vessel surveillance capsule at 17 EFPY based on the analysis of the reactor vessel surveillance capsule "T" which was previously withdrawn.

A copy of our related Safety Evaluation is also enclosed. A Notice of Issuance will be included in the Commission's next regular biweekly Federal Register notice.

/S/
Morton B. Fairtile, Project Manager
Project Directorate #1
Division of PWR Licensing-A

Enclosures:
1. Amendment No.15to
   License No. DPR-18
2. Safety Evaluation

cc w/enclosures:
See next page

*SEE PREVIOUS CONCURRENCE

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R. E. Ginna Nuclear Power Plant

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

ROCHESTER GAS AND ELECTRIC CORPORATION

DOCKET NO. 50-244

R. E. GINNA NUCLEAR POWER PLANT

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 15
License No. DPR-18

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Rochester Gas and Electric Corporation (the licensee) dated December 8, 1982 and supplemented October 10, 1983, August 8, 1984 and August 19, 1985 complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and Paragraph 2.C(2) of Facility Operating License No. DPR-18 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 15 , are hereby incorporated in the license.

The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Morton B. Fairtile

Morton B. Fairtile, Project Manager
Project Directorate #1
Division of PWR Licensing-A

Attachment:
Changes to the Technical
Specifications

Date of Issuance: June 12, 1986

ATTACHMENT TO LICENSE AMENDMENT NO. 15

FACILITY OPERATING LICENSE NO. DPR-18

DOCKET NO. 50-244

Revise Appendix A Technical Specifications by removing the pages identified below and inserting the enclosed pages. The revised pages are identified by the captioned amendment number and contain marginal lines indicating the area of change.

REMOVE	INSERT
3.1-10 to 3.1-16	3.1-10 to 3.1-16
--	3.1-16a
3.1-17 and 3.1-18	3.1-17 and 3.1-18
4.3-1	4.3-1
--	4.3-1a

3.1.2 Heatup and Cooldown Limit Curves for Normal Operation

3.1.2.1 The reactor coolant temperature and pressure and system heatup and cooldown rates (with the exception of the pressurizer) shall be limited in accordance with Figures 3.1-1 and 3.1-2 for the first 21.0 effective full power years.

a. Allowable combinations of pressure and temperature for specific temperature change rates are below and to the right of the limit lines shown. The heatup and cooldown rates shall not exceed 60°F/hr and 100°F/hr, respectively. Limit lines for cooldown rates between those presented may be obtained by interpolation.

b. Figures 3.1-1 and 3.1-2 define limits to assure prevention of non-ductile failure only. The limit lines shown in Figures 3.1-1 and 3.1-2 shall be recalculated periodically using methods discussed in the Basis Section.

3.1.2.2 The secondary side of the steam generator must not be pressurized above 200 psig if the temperature of the steam generator vessel is below 70°F.

3.1.2.3 The pressurizer heatup and cooldown rates shall not exceed 100°F/hr and 200°F/hr, respectively. The spray shall not be used if the temperature difference between the pressurizer and the spray fluid is greater than 320°F.

Basis: Fracture Toughness Properties

The fracture toughness properties of the ferritic materials in the reactor vessel are determined in accordance with the Summer 1965 Section III of the ASME Boiler and Pressure Vessel Code, Reference (1), and ASTM E185, Reference (2), and in accordance with additional reactor vessel requirements. These properties are then evaluated in accordance with Appendix G of the 1972 Summer Addenda to Section III of the ASME Boiler and Pressure Vessel Code, Reference (3) and the calculation methods described in Reference (4).

Heatup and cooldown limit curves are calculated using the most limiting value of RT_{NDT} at the end of 21.0 effective full power years (EFPY). The 21.0 EFPY period is chosen such that the limiting RT_{NDT} at the 1/4 T location in the core region is higher than the RT_{NDT} of the limiting unirradiated material. This service period assures that all components in the Reactor Coolant System will be operated conservatively in accordance with Code recommendations.

The highest RT_{NDT} of the core region material is determined by adding the radiation induced ΔRT_{NDT} for the applicable time period to the original RT_{NDT} shown in Reference (5). The fast neutron (E>1Mev) fluence at 1/4 thickness and 3/4 thickness vessel locations is given as a function of full power service life in Reference (5) and (6). Using the applicable fluence at the end of the 21.0 EFPY period for 1/4 thickness and the copper content of the material in question, the ΔRT_{NDT} is obtained from Reference (5). The ΔRT_{NDT} is more conservative than the value obtained from the third capsule of the radiation surveillance program.

Values of ΔRT_{NDT} determined in this manner will be used until more results from the material surveillance program, when evaluated according to ASTM E185, are available. The next capsule will be removed at approximately 17 EFPY (see Technical Specification 4.3.1). The heatup and cooldown curves must be recalculated when the ΔRT_{NDT} determined from the surveillance capsule is greater than the calculated ΔRT_{NDT} for the equivalent capsule radiation exposure.

Heatup and Cooldown Limit Curves

Allowable pressure temperature relationships for various heatup and cooldown rates are calculated using methods derived from Non-Mandatory Appendix G in Section III of the ASME Boiler and Pressure Vessel Code and discussed in detail in Reference (4). The approach specifies that the allowable total stress intensity factor (K_I) at any time during heatup or cooldown cannot be greater than that shown in the K_{IR}

curve for the metal temperature at that time. Furthermore, the approach applies explicit safety factors of 2.0 and 1.25* on stress intensity factors induced by pressure and thermal gradients, respectively. Thus, the governing equation for the heatup-cooldown analysis is:

(1) $2\ K_{Im} + 1.25\ K_{It} \leq K_{IR}$

where: K_{Im} is the stress intensity factor caused by membrane (pressure) stress.

K_{It} is the stress intensity factor caused by the thermal gradients.

K_{IR} is provided by the Code as a function of temperature relative to the RT_{NDT} of the material.

During the heatup analysis, Equation (1) is evaluated for two distinct situations.

First, allowable pressure-temperature relationships are developed for steady state (i.e., zero rate of

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* The 1.25 safety factor on K_{It} represents additional conservatism above Code requirements.

change of temperature) conditions assuming the presence of the code reference 1/4 T deep flaw at the ID of the pressure vessel. Due to the fact that during heatup the thermal gradients in the vessel wall tend to produce compressive stresses at the 1/4 T location, the tensile stresses induced by internal pressure are somewhat alleviated. Thus, a pressure-temperature curve based on steady state conditions (i.e., no thermal stresses) represents a lower bound of all similar curves for finite heatup rates when the 1/4 T location is treated as the governing factor.

The second portion of the heatup analysis concerns the calculation of pressure temperature limitations for the case in which the 3/4 T location becomes the controlling factor. Unlike the situation at the 1/4 T location, at the 3/4 T position (i.e., the tip of the 1/4 T deep OD flaw) the thermal gradients established during heatup produce stresses which are tensile in nature; and, thus, tend to reinforce the pressure stresses present. These thermal stresses are, of course, dependent on both the rate of heatup and the time (or water temperature) along the heatup ramp. Furthermore, since the thermal stresses at 3/4 T are tensile and increase with increasing heatup rate, a lower bound curve similar to that described in the preceding paragraph cannot be defined. Rather, each heatup rate of interest must be analyzed on an individual basis.

Following the generation of pressure-temperature curves for both the steady state and finite heatup rate situations, the final limit curves are produced in the following fashion. First, a composite curve is constructed based on a point by point comparison of the steady state and finite heatup rate data. At any given temperature, the allowable pressure is taken to be the lesser of the two values taken from the curves under construction. The composite curve is then adjusted to allow for possible errors in the pressure and temperature sensing instruments.

The use of the composite curve is mandatory in setting heatup limitations because it is possible for conditions to exist such that over the course of the heatup ramp the controlling analysis switches from the OD to the ID location; and the pressure limit must, at all times, be based on the most conservative case.

The cooldown analysis proceeds in the same fashion as that for heatup with the exception that the controlling location is always at the ID. The thermal gradients induced during cooldown tend to produce tensile stresses at the ID position and compressive stresses at the OD position. Thus, the ID flaw is clearly the worst case.

As in the case of heatup, allowable pressure-temperature relations are generated for both steady state and finite cooldown rate situations. Composite limit curves are then constructed for each cooldown rate of interest. Again adjustments are made to account for pressure and temperature instrumentation error.

The use of the composite curve in the cooldown analysis is necessary because system control is based on a measurement of reactor coolant temperature; whereas, the limiting pressure is calculated using the material temperature at the tip of the assumed reference flaw. During cooldown, the 1/4 T vessel location is at a higher temperature than the fluid adjacent to the vessel ID. This condition is, of course, not true for the steady state situation. It follows that the ΔT induced during cooldown results in a calculated higher K_{IR} for finite cooldown rates than for steady state under certain conditions.

Because operation control is on coolant temperature and cooldown rate may vary during the cooldown transient, the limit curves shown in Figure 3.1-2 represent a composite curve consisting of the more conservative values calculated for steady state and the specific cooling rate shown.

Details of these calculations are provided in Reference (4).

The temperature requirement for the steam generator corresponds with the measured NDT for the shell of the steam generator.

A temperature difference of 320°F between the pressurizer and reactor coolant system maintains thermal stresses within the pressurizer spray nozzle below design limits.

(1) ASME Boiler and Pressure Vessel Code Section III (Summer 1965)

(2) ASTM E185 Surveillance Tests on Structural Materials in Nuclear Reactors

(3) ASME Boiler and Pressure Vessel Code, Section III, Summer 1972 Addenda (note Code Class 1514)

(4) W.S. Hazelton, S.L. Anderson, and S.E. Yanichko, WCAP-7924, "Basis for Heatup and Cooldown Limit Curves"

(5) Analysis of Capsule T from the Rochester Gas and Electric Corporation R.E. Ginna Nuclear Plant Reactor Vessel Radiation Surveillance Program (WCAP-10086)

(6) Letter, R.W. Kober, RG&E to J.A. Zwolinski, USNRC, August 19, 1985

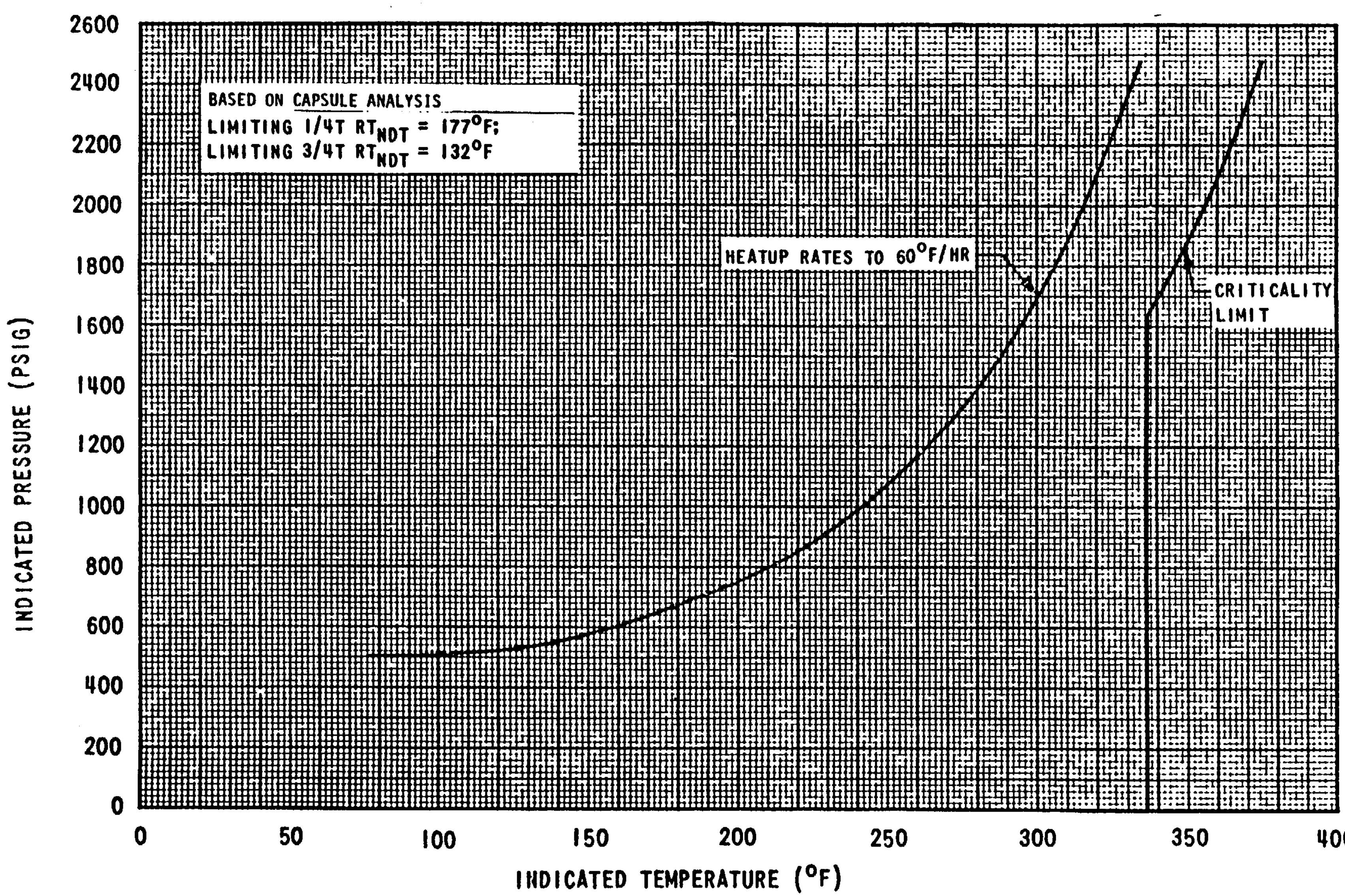

Figure 3.1-1 Reactor Coolant System Heatup Limitations Applicable for 21.0 Effective Full Power Years. $T_{ERROR} = 10^{\circ}F$, P_{ERROR} = 60 PSI

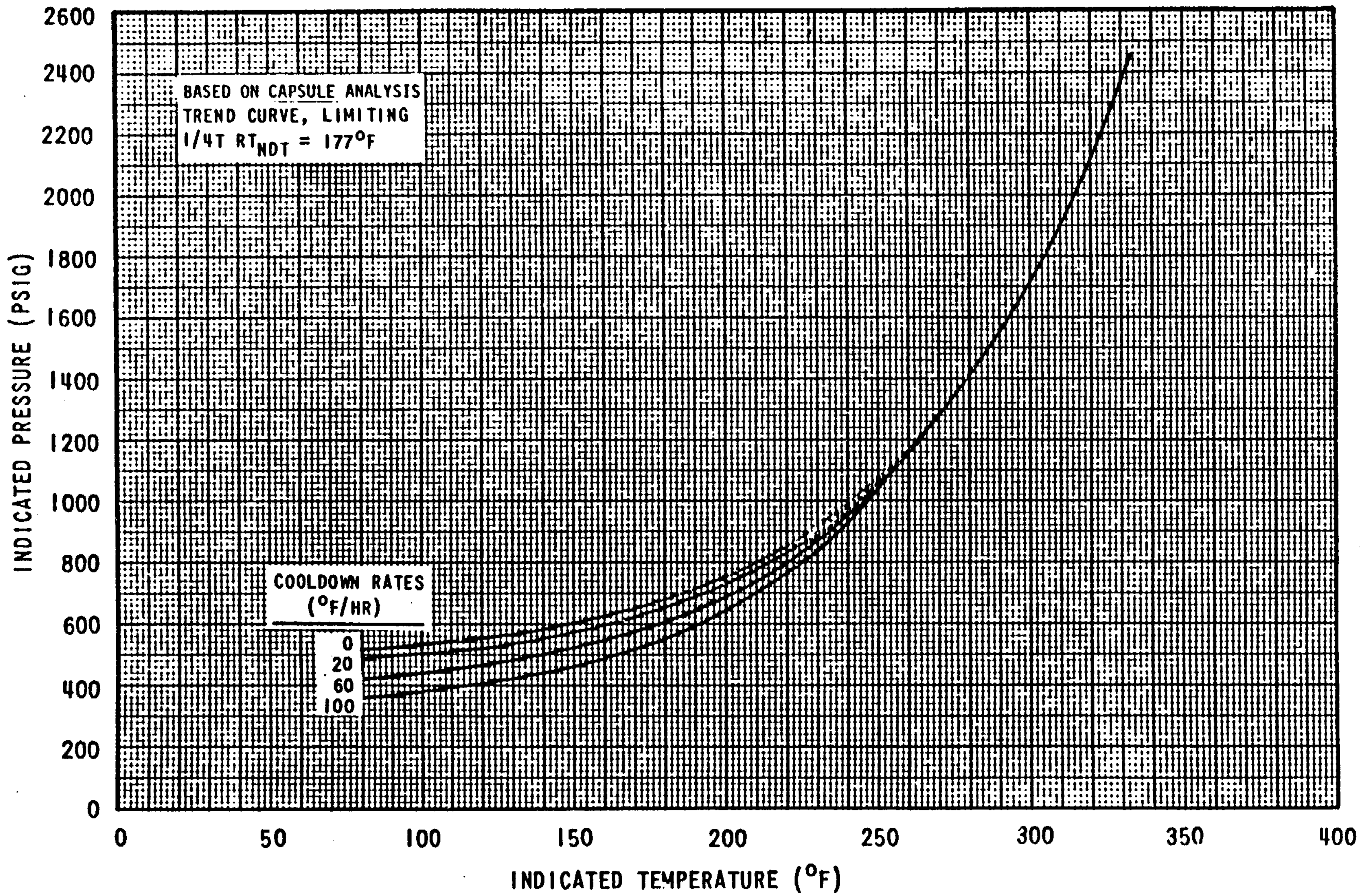

Figure 3.1-2 Reactor Coolant System Cooldown Limitations Applicable for 21.0 Effective Full Power Years. T_{ERROR} = 10°F, P_{ERROR} = 60 PSI

4.3 Reactor Coolant System

Applicability

Applies to surveillance of the reactor coolant system and its components.

Objective

To ensure operability of the reactor coolant system and its components.

Specifications:

4.3.1 Reactor Vessel Material Surveillance Testing

4.3.1.1 The reactor vessel material surveillance testing program is designed to meet the requirements of Appendix H to 10 CFR Part 50. This program consists of the metallurgical specimens receiving the following test: tensile, charpy impact and the WOL test. These tests of the Radiation Capsule Specimens shall be performed as follows:

Capsule	Time Removed For Testing
V	(Removed in 1971)
R	(Removed in 1974)
T	(Removed in 1980)
P	17 EFPY at nearest refueling
S	Standby
N	Standby

4.3.1.2 The report of the Reactor Vessel Material Surveillance shall be written as a Summary Technical Report as required by Appendix H to 10CFR Part 50.

4.3.2 Pressurizer

4.3.2.1 The pressurizer water level shall be verified to be within its limits at least once per 12 hours during power operation and hot shutdown.

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 ($^{n}/cm^2 x10^{19}$)	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:
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B. Elliot
M. Fairtile

Dated: June 12, 1986