UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

In the Matter of Rochester Gas and Electric Corporation (R.E. Ginna Nuclear Power Plant)	)))))	Docket No. 50-244

APPLICATION FOR AMENDMENT
TO OPERATING LICENSE

Pursuant to Section 50.90 of the regulations of the U.S. Nuclear Regulatory Commission (the "Commission"), Rochester Gas and Electric Corporation ("RG&E"), holder of Facility Operating License No. DPR-18, hereby requests that the Technical Specifications set forth in Appendix A to that license be amended. This request for change in Technical Specifications is to increase allowable reactor coolant activity levels to the Improved Technical Specification values (NUREG-1431).

A description of the amendment request, necessary background information, justification of the requested change, safety evaluation and no significant hazards and environmental considerations are provided in Attachment A. A marked up copy of the current Ginna Station Technical Specifications which shows the requested change is set forth in Attachment B. The proposed revised Technical Specifications are provided in Attachment C. These changes are consistent with Westinghouse Improved Technical Specifications (NUREG 1431) 3.4.16.a,b and figure 3.4.16-1.

WHEREFORE, Applicant respectfully requests that Appendix A to Facility Operating License No. DPR-18 be amended in the form attached hereto as Attachment C.

Rochester Gas and Electric Corporation

By Robert C. Mecredy
Robert C. Mecredy
Vice President
Ginna Nuclear Production

Subscribed and sworn to before me
on this 23rd day of May, 1994.

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ATTACHMENT A
R. E. GINNA POWER PLANT
LICENSE AMENDMENT REQUEST
TECHNICAL SPECIFICATION 3.1.4, MAXIMUM COOLANT ACTIVITY

This attachment provides a description of the amendment request and necessary justification for the proposed changes. The attachment is divided into seven sections as follows. Section A identifies all changes to the current Ginna Station Technical Specifications while Section B provides the background and history associated with the changes being requested. Section C provides detailed justification for the proposed changes including a comparison to Improved Technical Specifications as applicable. A safety evaluation, significant hazards consideration evaluation, and environmental consideration of the requested changes are provided in Sections D, E, and F, respectively. Section G lists all references used in this attachment.

A. Description of Amendment Request

This License Amendment Request (LAR) proposes to revise Ginna Station Technical Specifications 3.1.4.1.a, 3.1.4.1.b, figure 3.1.4-1 and associated Bases as follows:

1. Technical Specification 3.1.4.1.a

   i. The requirement is changed to "The total specific activity of the reactor coolant shall not exceed $100/\bar{E}$ μCi/gm,..."

   ii. The bases are revised to change the referenced analysis (Reference 3) to "UFSAR Section 15.6.3."

2. Technical Specification 3.1.4.1.b

   i. The requirement is revised to "The I-131 dose equivalent of the iodine activity in the reactor coolant shall not exceed 1.0 μCi/gm."

   ii. The bases are revised to change the referenced analysis (Reference 3) to "UFSAR Section 15.6.3."

3. Technical Specification Figure 3.1.4-1

   i. The allowable operation region is modified consistent with Improved Technical Specifications (see Attachments B and C for revised figure).

   ii. The bases are revised to change the referenced analysis (Reference 3) to "UFSAR Section 15.6.3."

B. Background

1. History

Prior to the January 25, 1982, steam generator tube rupture event at Ginna Station, reactor coolant activity limits were based on the original (1969) steam generator tube rupture analysis for the Ginna Station. The Commission's review of the 1982 tube rupture incident resulted in the requirement for a revised steam generator tube rupture analysis. The staff required that this be completed within six months of the plant restart (NUREG-0916, Section 9.0), and imposed reduced allowable activity levels in the interim (Amendment No. 51 to Provisional Operating License No. DPR-18, May 22, 1982). A bounding analysis using these reduced allowable activity levels was performed in order to satisfy the six month requirement, while a more detailed analysis supporting the standard technical specification values would follow. The methodology for this new analysis (WCAP-10698-P-A) was submitted and approved by the Commission for use on Westinghouse PWRs provided five plant specific inputs were verified to be consistent with the assumptions in the methodology (Reference a).

RG&E has completed this verification, and therefore intends to update its analysis of record for the steam generator tube rupture to reflect use of this new methodology (UFSAR Section 15.6.3). This new analysis supports the activity limits proposed in this Amendment.

2. Hardware Modifications

This LAR involves no hardware changes to Ginna Station.

C. Justification

This proposed Amendment imposes reactor coolant activity limits consistent with NUREG-1431, "Westinghouse Standard Technical Specifications." The applicability of these limits for Ginna Station are established by a plant specific steam generator tube rupture and radiological consequences analysis, WCAP-11668, which is consistent with the approved methodology of WCAP-10698-P-A for analysis of steam generator tube rupture transients. All contingencies for usage of WCAP-10698-P-A methodology (Reference a) have been satisfied for Ginna Station as described in section D below.

D. Safety Evaluation

Potential environmental consequences of a steam generator tube rupture event at the R.E. Ginna nuclear power plant have been evaluated to verify that the Improved Technical Specification limit on primary coolant activity is adequate for Ginna. This analysis, WCAP-11668 (attached) is consistent with the methodology described in WCAP-10698-P-A. The Commission requires that five contingencies be met in order to use this methodology, specifically:

1. Demonstration that critical operator action times used in the analysis are realistic and consistent with those observed during simulator exercises.

2. A site specific Steam Generator Tube Rupture radiological offsite consequence analysis.

3. A structural analysis of the main steam lines demonstrating adequacy under water-filled

conditions.

4. A list of systems, components, and instrumentation credited for accident mitigation and the specified safety grade for each.

5. A comparison of the plant to the "bounding plant" used in WCAP-10698.

Compliance with those contingencies for Ginna Station has been satisfied and is described below.

1. Demonstration that critical operator action times used in the analysis are realistic and consistent with those observed during simulator exercises.

During the week of August 19 through 23, 1991, simulator exercises were performed at the Ginna Station simulator to verify the assumptions used for both analyses cases presented in WCAP-11668. The results are tabulated below.

CASE 1, INTACT SG PORV FAILS CLOSED		
OPERATOR ACTION	WCAP 11668 TIME (SEC)	SIMULATOR TIME (SEC)
1. Recognize and Isolate Ruptured SG	600	423
2. Recognize and locally open intact SG PORV open	1804	1460*
3. Terminate SI	2798	1916
4. Terminate break flow	3428	2541

* The simulator exercise imposed a 15 min. delay from when the operator identified the failed PORV to when the PORV was locally opened to account for operator actions outside the control room which could not be verified on the simulator. This delay is consistent with the assumptions in WCAP-11668. Simulation of these actions in the actual plant have demonstrated that these times are conservative.

CASE 2 RUPTURED SG PORV FAILS OPEN		
OPERATOR ACTION	WCAP-11668 TIME (SEC)	SIMULATOR TIME (SEC)
1. Ruptured SG Isolated	652	214
2. Recognize and Locally Isolate Failed PORV	1558	1116*
3. Terminate SI	3066	2073
4. Terminate Break Flow	3438	2424

* The simulator exercise imposed a 15 min. delay from when the operator identified the failed PORV to when the PORV was locally isolated to account for operator actions outside the control room which could not be verified on the simulator. This delay is consistent with the assumptions in WCAP-11668. Simulation of these actions in the actual plant have demonstrated that these times are conservative.

These simulator exercises demonstrate that the critical operator action times assumed in WCAP-11668 are realistic and conservative and therefore this contingency is satisfied.

2. Provide a site specific Steam Generator Tube Rupture radiological offsite consequences analysis.

   WCAP-11668, provided with this LAR provides a Ginna site specific Steam Generator Tube Rupture radiation offsite consequences analysis, and therefore, this contingency is satisfied.

3. Provide a structural analysis of the main steam lines demonstrating adequacy under water-filled conditions.

   Prior to restart of Ginna Station following the January 25, 1982, tube rupture incident, a main steam line structural analysis under water-filled conditions was performed and provided to the Commission. The acceptability of this analysis is documented in the restart SER, NUREG-0916, section 6.0. Therefore, this contingency is met.

4. A list of systems, components, and instrumentation credited for accident mitigation and the specified safety grade for each.

   In response to NUREG-0737, Supplement 1 Item 6.2, RG&E has provided post accident instrumentation qualification information. A comprehensive table listing the credited equipment, its qualification, and all other attributes listed in Regulatory Guide 1.97, revision 3, was provided to the NRC by letter R. Mecredy to A. Johnson "Emergency

Response Capability", dated October 14, 1992. An SER for this submittal was provided to RG&E by letter A. Johnson to R. Mecredy, "Emergency Response Capability," dated February 24, 1993. Therefore, this contingency has been satisfied.

5. A comparison of the plant to the "bounding plant" used in WCAP-10698.

Plant parameters for the reference plant used in WCAP-10698-P-A are provided in Table 4.3-3 of the WCAP. WCAP-11668, the Ginna specific analysis, utilizes Ginna specific parameters. All Ginna specific parameters fall within the bounds of the parameters listed in WCAP-10698-P-A as detailed below:

PLANT PARAMETER	WCAP-10698 BASE CASE	WCAP-10698 CONSERVATIVE	WCAP-11668 GINNA
RCS Pressure, psia	2250	2220	2220
Pressurizer Water Volume, ft^3	750	868	800
SG Secondary Mass, lbm	107759	118535	103256
Reactor Trip Delay, sec	2.0	0.0	2.0
Turbine Trip Delay, sec	0.3	0.0	0.3
Pressurizer Pressure for SI, psia	1864	1889	1750
Pressurizer Pressure for Reactor Trip, psia	1960	1985	1902
SG Relieve Pressure, psia	1100	1050	1060
SIS Pump Delay, sec	12	0.0	0.0
AFW Delay, sec	60	0.0	0.0
AFW Flow Rate, gpm	1839	1839	800
AFW Temperature, °F	40	120	120
Decay Heat	100% ANS	120% ANS	120% ANS

It should be noted that the methodology of WCAP-10698-P-A provides a benchmark against the 1982 Ginna tube rupture incident, and, therefore, its applicability to Ginna is explicit.

Therefore, this contingency is satisfied.

Based on the above, the methodology described in WCAP-10698-P-A can be applied to Ginna. WCAP-11668 (enclosed) provides the results of this application, and demonstrates the acceptability of Improved Technical Specification coolant activity limits for Ginna.

Therefore, the proposed amendment does not involve an unreviewed safety question and will not adversely affect or endanger the health and safety of the general public.

E. Significant Hazards Consideration Evaluation

The proposed changes to the Ginna Station Technical Specifications do not involve a significant hazards consideration as discussed below:

1. Operation of Ginna Station in accordance with the proposed changes does not involve a significant increase in the probability or consequences of an accident previously evaluated.

   The proposed changes do not affect any accident initiators and therefore the probability of any accident is not increased. Consequences of the changes are analyzed and shown acceptable in the enclosed analysis, WCAP-11668, Section III.

2. Operation of Ginna Station in accordance with the proposed changes does not create the possibility of a new or different kind of accident from any accident previously evaluated.

   The proposed changes involve no physical modifications to the plant; therefore, no new accident can be postulated.

3. Operation of Ginna Station in accordance with the proposed changes does not involve a significant reduction in a margin of safety, as no margin of safety is reduced by the proposed changes, as shown in WCAP-11668.

   Based upon the above information, it has been determined that the proposed changes to the Ginna Station Technical Specifications do not involve a significant increase in the probability or consequences of an accident previously evaluated, does not create the possibility of a new or different kind of accident previously evaluated, and does not involve a significant reduction in a margin of safety. Therefore, it is concluded that the proposed changes meet the requirements of 10 CFR 50.92(c) and do not involve a significant hazards consideration.

F. Environmental Consideration

RG&E has evaluated the proposed changes and determined that:

1. The changes do not involve a significant hazards consideration as documented in Section E above;

2. The changes do not involve a significant change in the types or significant increase in the amounts of any effluents that may be released offsite as demonstrated in the enclosed analysis, WCAP 11668.

3. The changes do not involve a significant increase in individual or cumulative occupational radiation exposure since the change does not affect allowable limits.

Accordingly, the proposed changes meet the eligibility criteria for categorical exclusion set forth in 10 CFR

51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), an environmental assessment of the proposed changes is not required.

G. References

(a): NRC Letter, C. Rossi to A. Ladieu (WOG), "Acceptance for Referencing of Licensing Topical Report WCAP-10698...", March 30, 1987.

(b): NUREG-0916, "Safety Evaluation Report Related to the Restart of R.E. Ginna Nuclear Power Plant", May 1982.

(c): RG&E Letter, R. Mecredy to A. Johnson (NRC), "Emergency Response Capability...", October 14, 1992.

(d): NRC Letter, A. Johnson to R. Mecredy (RG&E), Emergency Response Capability - Conformance to Regulatory Guide 1.97, revision 3", February 24, 1993.