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

July 30, 1985

Docket No. 50-244
LS05-85-07-042

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

Dear Mr. Kober:

SUBJECT: TMI ACTION PLAN TECHNICAL SPECIFICATIONS

Re: R. E. Ginna Nuclear Power Plant

The Commission has issued the enclosed Amendment No. 9 to Facility Operating License No. DPR-18 for the R.E. Ginna Nuclear Power Plant. This amendment is in response to your application dated September 14, 1984 and superseded on February 21, 1985.

The amendment adds limiting conditions for operation (LCO) and Surveillance requirements to the Technical Specifications for various plant modifications required by TMI Action Plan Items covered by Generic Letter 83-37. These modifications are: (1) Reactor Coolant System vents (II.B.1); (2) Noble Gas Effluent Monitors (II.F.1.1); (3) Containment High-Range Radiation Monitor (II.F.1.3); (4) Containment Pressure Monitor (II.F.1.4); (5) Containment Water Level Monitor (II.F.1.5); (6) Containment Hydrogen Monitor (II.F.1.6); (7) Instrumentation for Detection of Inadequate Core Cooling (II.F.2); and (8) Control Room Habitability Requirements (III.D.3.4).

The staff has determined that the following items are covered by existing Technical Specifications:

1. Post Accident Sampling (II.B.3)
2. Long Term Auxiliary Feedwater System Evaluation (II.E.1.1)
3. Sampling and Analysis of Plant Effluents (II.F.1.2)

For Item II.F.2 the licensee has not completed the hardware modifications required for the reactor coolant inventory tracking system. This portion of Item II.F.2 will be addressed in future correspondence and is considered the sole item remaining open to be in full compliance with G.L. 83-37.

SE01 1/1 GX/07

A Notice of Consideration of Issuance of Amendment to License and Proposed No Significant Hazards Consideration Determination and Opportunity for Hearing related to the requested action was published in the Federal Register on May 21, 1985 (50 FR 20987). No public comments or requests for hearing were received.

A copy of our related Safety Evaluation is also enclosed. This action will appear in the Commission's biweekly notice publication in the Federal Register.

Sincerely,

Original signed by:

John A. Zwolinski, Chief
Operating Reactors Branch #5
Division of Licensing

Enclosures:
1. Amendment No. 9 to License No. DPR-18
2. Safety Evaluation

cc w/enclosures:
See next page

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Mr. Roger W. Kober
Rochester Gas and Electric Corporation

R. E. Ginna Nuclear Power Plant

cc
Harry H. Voigt, Esquire
LeBoeuf, Lamb, Leiby and MacRae
1333 New Hampshire Avenue, N.W.
Suite 1100
Washington, D.C. 20036

Ezra Bialik
Assistant Attorney General
Environmental Protection Bureau
New York State Department of Law
2 World Trade Center
New York, New York 10047

Resident Inspector
R.E. Ginna Plant
c/o U.S. NRC
1503 Lake Road
Ontario, New York 14519

Stanley B. Klimberg, Esquire
General Counsel
New York State Energy Office
Agency Building 2
Empire State Plaza
Albany, New York 12223

Regional Administrator
Nuclear Regulatory Commission
Region I Office
631 Park Avenue
King of Prussia, Pennsylvania 19406

Supervisor of the Town of Ontario
1850 Ridge Road
Ontario, New York 14519

Jay Dunkleberger
Division of Policy Analysis & Planning
New York State Energy Office
Agency Building 2
Empire State Plaza
Albany, New York 12223

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. 9
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 September 14, 1984 and superseded on February 21, 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) <u>Technical Specifications</u>

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 9 , 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

John A. Zwolinski, Chief
Operating Reactors Branch #5
Division of Licensing

Attachment:
Changes to the Technical
  Specifications

Date of Issuance: July 30, 1985.

ATTACHMENT TO LICENSE AMENDMENT NO. 9

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
3.1-6	3.1-6 and 3.1-6a
3.1-9	3.1-9 and 3.1-9a
3.5-1 through 3.5-3	3.5-1 through 3.5-3a
3.5-5	3.5-5 and 3.5-5a
3.5-15	3.5-15
--	3.5-22
3.6-2	3.6-2
4.1-7	4.1-7
4.1-10	4.1-10
4.1-12 and 4.1-13	4.1-12 and 4.1-13
4.3-5	4.3-5 and 4.3-6
4.4-11	4.4-11 and 4.4-11a

3.1.1.5 Pressurizer

Whenever the reactor is at hot shutdown or critical, the pressurizer shall have at least 100 kw of heaters operable and a water level maintained between 12% and 87% of level span. If the pressurizer is inoperable due to heaters or water level, restore the pressurizer to operable status within 6 hours or have the RHR system in operation within an additional 6 hours.

3.1.1.6 Reactor Coolant System Vents

a. When the reactor is at hot shutdown or critical, at least one reactor coolant system vent path consisting of two valves in series shall be operable and closed* at each of the following locations:

   1. Reactor Vessel head
   2. Pressurizer steam space

*The PORV block valve is not required to be closed but must be operable if the PORV is capable of being opened.

b. With one or more vents at the above reactor coolant system vent path locations inoperable, startup may commence and/or power operation may continue provided at least one vent path is operable and the inoperable vent paths are maintained closed with motive power removed from the valve actuator of all the valves in the inoperable vent paths. If the requirements of 3.1.1.6a are not met within 30 days, be in hot shutdown within 6 hours and below 350°F within the following 30 hours.

c. With all of the above reactor coolant system vent paths inoperable; maintain the inoperable vent paths closed with power removed from the valve actuators of all the valves in the inoperable vent paths, and restore at least one of the vent paths to operable status within 72 hours or be in hot shutdown within 6 hours and below 350°F within the following 30 hours.

Bases

The plant is designed to operate with all reactor coolant loops in operation and maintain the DNBR above the limit value during all normal

Temperature requirements for the steam generator correspond with measured NDT for the shell and allowable thermal stresses in the tube sheet.

Each of the pressurizer code safety valves is designed to relieve 288,000 lbs. per hr. of saturated steam at the valve set point. Below 350°F and 350 psig in the reactor coolant system, the residual heat removal system can remove decay heat and thereby control system temperature and pressure. If no residual heat were removed by any of the means available, the amount of steam which could be generated at safety valve relief pressure would be less than half the valves' capacity. One valve, therefore, provides adequate defense against overpressurization.

The power operated relief valves (PORVs) operate to relieve RCS pressure below the setting of the pressurizer code safety valves. These relief valves have remotely operated block valves to provide a positive shutoff capability should a relief valve become inoperable. The electrical power for both the relief valves and the block valves is capable of being supplied from an emergency power source to ensure the ability to seal this possible RCS leakage path. The requirement that 100 kw of pressurizer heaters and their associated controls be capable of being supplied electrical power from an emergency bus provides assurance that these heaters can be energized during a loss of offsite power condition to maintain natural circulation at hot shutdown and during cooldown.[3]

Reactor Coolant System Vents

Reactor Coolant System Vents are provided to exhaust noncondensible gases and/or steam from the primary system that could inhibit natural circulation core cooling. The operability of at least one reactor coolant system vent path from the reactor vessel head and one from the pressurizer steam space ensures the capability exists to perform this function.

The valve redundancy of the reactor coolant system vent paths serves to minimize the probability of inadvertent or irreversible actuation while ensuring that a single failure of a vent valve, power supply or control system does not prevent isolation of the vent path.

The function, capabilities, and testing requirements of the reactor coolant system vent systems are consistent with the requirements of Item II.B.1 of NUREG-0737, "Clarification of TMI Action Plan Requirements", November 1980.

References

(1) FSAR Section 14.1.6
(2) FSAR Section 7.2.3
(3) Letter from L.D. White, Jr. to D. L. Ziemann, USNRC, dated October 17, 1979

3.5 Instrumentation Systems

Applicability:

Applies to plant instrumentation systems.

Objective:

To delineate the conditions of the plant instrumentation and safety circuits and to limit the release of radioactive materials.

Specification:

3.5.1 Operational Safety Instrumentation

3.5.1.1 The number of Minimum Operable Channels for instrumentation shown on Tables 3.5-1 through 3.5-3 shall be OPERABLE for plant operation at rated power.

3.5.1.2 In the event the number of channels of a particular subsystem in service falls below the limit given in the columns entitled Minimum Operable Channels, operation shall be limited according to the requirement shown in the last column of Tables 3.5-1 through 3.5-3.

3.5.2 Accident Monitoring Instrumentation

3.5.2.1 The accident monitoring instrumentation channels shown in Table 3.5-4 shall be operable whenever the reactor is at hot shutdown or is critical.

3.5.2.2 While critical, with the number of operable accident monitoring instrumentation channels less than the Total Required Number of Channels shown in Table 3.5-4, either restore the inoperable channel(s) to operable status within 7 days, or be in at least hot shutdown within the next 12 hours.

3.5.2.3 While critical, with the number of operable accident monitoring instrumentation channels less than the MINIMUM CHANNELS OPERABLE requirements of Table 3.5-4, either restore the inoperable channel(s) to operable status within 48 hours or be in at least hot shutdown within the next 12 hours.

3.5.2.4 The radiation accident monitoring instrumentation channels shown in Table 3.5-7 shall be operable, whenever the reactor is at hot shutdown or is critical. With one or more radiation monitoring channels inoperable, take the action shown in Table 3.5-7. Startup may commence or continue consistent with the action statement.

3.5.3 Engineered Safety Feature Actuation Instrumentation

3.5.3.1 The Engineered Safety Feature Actuation System (ESFAS) instrumentation channels shown in Tables 3.5-2 and 3.5-3 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.5-5.

3.5.3.2 With an instrumentation channel trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.5-5, declare the channel inoperable and apply the applicable ACTION requirement of Tables 3.5-2 and 3.5-3 until the channel is restored to OPERABLE status with the trip setpoint adjusted consistent with the Trip Setpoint Value.

3.5.3.3 With an instrumentation channel inoperable, take the action shown in Tables 3.5-2 and 3.5-3.

3.5.4 Radioactive Effluent Monitoring Instrumentation

3.5.4.1 The radioactive effluent monitoring instrumentation shown in Table 3.5-6 shall be operable at all times with alarm and/or trip setpoints set to ensure that the limits of Specifications 3.9.1.1 and 3.9.2.1 are not exceeded. Alarm and/or trip setpoints shall be established in accordance with calculational methods set forth in the Offsite Dose Calculation Manual.

3.5.4.2 If the setpoint for a radioactive effluent monitor alarm and/or trip is found to be higher than required, one of the following three measures shall be taken immediately:

(i) the setpoint shall be immediately corrected without declaring the channel inoperable; <u>or</u>

(ii) immediately suspend the release of effluents monitored by the affected channel; <u>or</u>

(iii) declare the channel inoperable.

3.5.4.3 If the number of channels which are operable is found to be less than required, take the action shown in Table 3.5-6.

3.5.5 <u>Control Room HVAC Detection Systems</u>

3.5.5.1 During all modes of plant operation, detection systems for chlorine gas, ammonia gas and radioactivity in the control room HVAC intake shall be operable with setpoints to isolate air intake adjusted as follows:

chlorine, $\leq$ 5 ppm

ammonia, $\leq$ 35 mg/m^3

radioactivity, particulate $\leq 1 \times 10^{-8}$ uc/cc

iodine $\leq 9 \times 10^{-9}$ uc/cc

noble gas $\leq 1 \times 10^{-5}$ uc/cc

3.5.5.2 With one of the detection systems inoperable, within 1 hour isolate the control room HVAC air intake. Maintain the air intake isolated except for short periods, not to exceed 1 hour a day, when fresh air makeup is allowed to improve the working environment in the control room.

Basis

During plant operations, the complete instrumentation systems will normally be in service. Reactor safety is provided by the Reactor Protection System, which automatically initiates appropriate action to prevent exceeding established limits. Safety is not compromised, however, by continuing operation with certain instrumentation channels out of service since provisions were made for this in the plant design. This specification outlines limiting conditions for operation necessary to preserve the effectiveness of the reactor control and protection system when any one or more of the channels is out of service.

The radioactive gaseous effluent instrumentation is provided to monitor and control, as applicable, the releases of radioactive materials in gaseous effluents. The alarm and/or trip setpoints for these instruments are calculated in accordance with the ODCM to ensure that alarm and/or trip will occur prior to exceeding the limits of 10 CFR Part 20. This instrumentation also includes provisions for monitoring the concentrations of potentially explosive gas mixtures in the waste gas holdup system. The operability and use of this instrumentation is consistent with the requirements of General Design Criterion 64 of Appendix A to 10 CFR Part 50. Control room HVAC detection systems are designed to prevent the intake of chlorine, ammonia and radiation at concentrations which may prevent plant operators from performing their required functions. Concentrations which initiate isolation of the control room HVAC system have been established using the guidance of several established references (2-4).

The chlorine isolation setpoint is 1/3 of the toxicity limit of reference 2 but slightly greater than the short term exposure limit of reference 4. The ammonia setpoint is established at approximately 1/3 of the toxicity limit for anhydrous ammonia in reference 2 and equal to the short term exposure limit of reference 4. The setpoints for radioactivity correspond to the maximum permissible concentrations of reference 3 for Cs-137, I-131 and Kr-85.

References

1) Updated FSAR - Section 7.2.

2) USNRC Regulatory Guide 1.78, June 1974, Assumptions for Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release.

3) 10 CFR 20 Appendix B, Table I.

4) Threshold Limit Values for Chemical Substances and Physical Agents in the Work Environment, 1982. Published by American Conference of Governmental Industrial Hygienists.

TABLE 3.5-4

Accident Monitoring Instrumentation

INSTRUMENT	TOTAL REQUIRED NO. OF CHANNELS (7)	MINIMUM CHANNELS OPERABLE (7)
1. Pressurizer Water Level (1)	2	1
2. Auxiliary Feedwater Flow Rate (2)(3)	2/steam generator	1/steam generator
3. Steam Generator Water Level - Wide Range (3)	1/steam generator	1/steam generator
4. Reactor Coolant System Subcooling Margin Monitor (4)	2	1
5. Pressurizer PORV Position Indicator (5)	2/Valve	1/Valve
6. PORV Block Valve Position Indicator (1)	1/Valve	0/Valve
7. Pressurizer Safety Valve Position Indicator (5)	2/Valve	1/Valve
8. Containment Pressure (8)	2	1
9. Containment Water Level (Narrow Range, Sump A)	1(6)	1(6)
10. Containment Water Level (Wide Range, Sump B)	2	1
11. Core-Exit Thermocouples	4/core quadrant	2/core quadrant

Notes

(1) Emergency power for pressurizer equipment, NUREG-0737, item II.G.1.

(2) Auxiliary feedwater system flow indication, NUREG-0737, item II.E.1.2.

(3) Only 2 out of the 3 indications (two steam generator auxiliary feedwater flow and one wide-range steam generator level) are required to be operable, NUREG-0737, item II.E.1.2.

(4) Instrumentation for detection of inadequate core cooling, NUREG-0737, item item II.F.2.1.

(5) Direct indication of relief and safety valve position, NUREG-0737, item II.D.3. Two channels include a primary detector and RTD as the backup detector.

(6) Operation may continue with less than the minimum channels operable provided that the requirements of Technical Specification 3.1.5.3 are met.

(7) See Specification 3.5.2 for required action.

(8) Containment pressure monitor, NUREG-0737, item II.F.1.4.

TABLE 3.5-7

Radiation Accident Monitoring Instrumentation

Instrument	Minimum Channels Operable	Action
1. Containment Area (R-29 and R-30)	2	1
2. Noble Gas Effluent Monitors		
i. Plant Vent (R-14A)	1	1
ii. A Main Steam Line (R-31)	1	1
iii. B Main Steam Line (R-32)	1	1
iv. Containment Purge (R-12A)	1	1
v. Air Ejector (R-15A)	1	1

Action Statements

Action 1 - With the number of operable channels less than required by the Minimum Channels Operable requirements, either restore the inoperable channel(s) to operable status within 7 days of the event, or prepare and submit a Special Report to the Commission within 30 days following the event outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to operable status.

3.6.3 Containment Isolation Valves

3.6.3.1 With one or more of the isolation valve(s) specified in Table 3.6-1 inoperable, maintain at least one isolation valve operable in each affected penetration that is open and either:

a. Restore the inoperable valve(s) to operable status within 4 hours, or

b. Isolate each affected penetration within 4 hours by use of at least one deactivated automatic valve secured in the isolation position, or

c. Isolate each affected penetration within 4 hours by use of at least one closed manual valve or blind flange, or

d. Be in at least hot shutdown within the next 6 hours and in cold shutdown within the following 30 hours.

Isolation valves are inoperable from a leakage standpoint if the leakage is greater than that allowed by 10 CFR 50 Appendix J.

3.6.4 Combustible Gas Control

3.6.4.1 When the reactor is critical, at least two independent containment hydrogen monitors shall be operable. One of the monitors may be the Post Accident Sampling System.

3.6.4.2 With only one hydrogen monitor operable, restore a second monitor to operable status within 30 days or be in at least hot shutdown within the next 6 hours.

3.6.4.3 With no hydrogen monitors operable, restore at least one monitor to operable status within 72 hours or be in at least hot shutdown within the next 6 hours.

TABLE 4.1-1 (CONTINUED)

	Channel Description	Check	Calibrate	Test	Remarks
25.	Containment Pressure	S	R	M	Narrow range containment pressure (-3.0, +3 psig) excluded
26.	Steam Generator Pressure	S	R	M	
27.	Turbine First Stage Pressure	S	R	M	
28.	Emergency Plan Radiation Instruments	M	R	M	
29.	Environmental Monitors	M	NA	NA	
30.	Loss of Voltage/Degraded Voltage 480 Volt Safeguards Bus	NA	R	M	
31.	Trip of Main Feedwater Pumps	NA	NA	R	
32.	Steam Flow	S	R	M	
33.	T_{AVG}	S	R	M	
34.	Chlorine Detector, Control Room Air Intake	NA	R	M	
35.	Ammonia Detector, Control Room Air Intake	NA	R	M	
36.	Radiation Detectors, Control Room Air Intake	NA	R	M	

TABLE 4.1-3

Accident Monitoring Instrumentation Surveillance Requirements

Instrument	Channel Check	Channel Calibration	Channel Test
1. Pressurizer Water Level (1)	see Table 4.1-1	see Table 4.1-1	NA
2. Auxiliary Feedwater Flow Rate (4)	see Section 4.8.1	R	NA
3. Reactor Coolant System Subcooling Margin Monitor (2)	M	R	NA
4. Pressurizer PORV Position Indicator (primary detector) (3)	M	NA	R
5. Pressurizer PORV Position Indicator (RTD - backup detector) (3)	M	R	NA
6. PORV Block Valve Position Indicator (1)	M	NA	R
7. Pressurizer Safety Valve Position Indicator (primary detector) (3)	M	R	NA
8. Presurizer Safety Valve Position Indicator (RTD - backup detector) (3)	M	R	NA
9. Containment Pressure	M	R	NA
10. Steam Generator Water Level - Wide Range	M	R	NA
11. Containment Water Level (Narrow Range, Sump A)	M	R	NA
12. Containment Water Level (Wide Range, Sump B)	M	R	NA
13. Core Exit Thermocouples	M	R	NA
14. Containment Area High Range Radiation (R-29 and R-30) (5)	M	R	M

(1) Emergency Power Supply Requirements for Pressurizer Level Indicators - NUREG 0578 Item 2.1.1
(2) Instrumentation for Detection of Inadequate Core Cooling - NUREG 0578 Item 2.1.1
(3) Direct Indication of Power Operated Relief Valve and Safety Valve Position - NUREG 0578 item 2.1.3.a
(4) Auxiliary Feedwater Flow Indication to Steam Generator NUREG 0578 item 2.1.7.b
(5) Acceptable criteria for calibration are provided in Table II.F.1-3 of NUREG 0737

TABLE 4.1-5

Radioactive Effluent Monitoring Surveillance Requirements

Instrument	Channel Check	Source Check	Functional Test	Channel Calibration
1. Gross Activity Monitor (Liquid)				
a. Liquid Rad Waste (R-18)	D(7)	M(4)	Q(1)	R(5)
b. Steam Generator Blowdown (R-19)	D(7)	M(4)	Q(1)	R(5)
c. Turbine Building Floor Drains (R-21)	D(7)	M(4)	Q(1)	R(5)
d. High Conductivity Waste (R-22)	D(7)	M(4)	Q(1)	R(5)
e. Containment Fan Coolers (R-16)	D(7)	M(4)	Q(2)	R(5)
f. Spent Fuel Pool Heat Exchanger (R-20)	D(7)	M(4)	Q(2)	R(5)
2. Plant Ventilation				
a. Noble Gas Activity (R-14) (Alarm and Isolation of Gas Decay Tanks)	D(7)	M	Q(1)	R(5)
b. Particulate Sampler (R-13)	W(7)	N.A.	N.A.	R(5)
c. Iodine Sampler (R-10B and R-14A)	W(7)	N.A.	M	R(5)
d. Flow Rate Determination	N.A.	N.A.	N.A.	R(6)
3. Containment Purge				
a. Noble Gas Activity (R-12)	D(7)	PR	Q(1)	R(5)
b. Particulate Sampler (R-11)	W(7)	N.A.	Q(1)	R(5)
c. Iodine Sampler (R-10A and R-12A)	W(7)	N.A.	M	R(5)
d. Flow Rate Determination	N.A.	N.A.	N.A.	R(6)
4. Air Ejector Monitor (R-15 and R-15A)	D(7)	M	M(2)	R(5)
5. Waste Gas System Oxygen Monitor	D	N.A.	N.A.	Q(3)
6. Main Steam Lines (R-31 and R-32)	M	N.A.	Q	R

TABLE 4.1-5 (Continued)

TABLE NOTATION

(1) The Channel Functional Test shall also demonstrate that automatic isolation of this pathway and control room alarm occur if any of the following conditions exist:

1. Instrument indicates measured levels above the alarm and/or trip setpoint.

2. Power failure.

(2) The Channel Functional Test shall also demonstrate that control room alarm occurs if any of the following conditions exist:

1. Instrument indicates measured levels above the alarm setpoint.

2. Power failure.

(3) The Channel Calibration shall include the use of standard gas samples containing a nominal:

1. Zero volume percent oxygen; and

2. Three volume percent oxygen.

(4) This check may require the use of an external source due to high background in the sample chamber.

(5) Source used for the Channel Calibration shall be traceable to the National Bureau of Standards (NBS) or shall be obtained from suppliers (e.g. Amersham) that provide sources traceable to other officially-designated standards agencies.

(6) Flow rate for main plant ventilation exhaust and containment purge exhaust are calculated by the flow capacity of ventilation exhaust fans in service and shall be determined at the frequency specified.

(7) Applies only during releases via this pathway.

a) to demonstrate a reactor coolant loop operable, the reactor coolant pump(s), if not in operation, shall be demonstrated operable at least once per 7 days by verifying correct breaker alignments and indicated power availability.

b) to demonstrate a residual heat removal pump is operable, the surveillance specified in the Inservice Pump and Valve Test Program prepared pursuant to 10 CFR 50.55a shall be performed.

4.3.5.4 When the reactor is at cold shutdown or when the average coolant temperature is between 200°F and 350°F and fuel is in the reactor, at least one coolant loop shall be verified to be in operation and circulating reactor coolant at least once per 12 hours.

4.3.5.5 In addition to the above requirements, in order to demonstrate that a reactor coolant loop is operable, the steam generator water level shall be greater than or equal to 16% of the narrow range instrument span.

4.3.5.6 Each reactor coolant system vent path shall be demonstrated operable at least once per 18 months by:

1. Verifying all manual isolation valves in each vent path are locked in the open position.
2. Verifying flow through the reactor coolant vent system vent paths using either liquid or gas.

Basis:

This material surveillance program monitors changes in the fracture toughness properties of ferritic materials in the reactor vessel beltline region of the reactor resulting from exposure to neutron irradiation and the thermal environment. The test data obtained from this program will be used to determine the conditions under which the reactor vessel can be operated with adequate margins of safety against fracture throughout its service life.

The surveillance requirements on pressurizer equipment will assure proper performance of the pressurizer function and give early indication of malfunctions.

the tendon containing 6 broken wires) shall be inspected. The acceptance criterion then shall be no more than 4 broken wires in any of the additional 4 tendons. If this criterion is not satisfied, all of the tendons shall be inspected and if more than 5% of the total wires are broken, the reactor shall be shut down and depressurized.

4.4.4.2 Pre-Stress Confirmation Test

a. Lift-off tests shall be performed on the 14 tendons identified in 4.4.4.1a above, at the intervals specified in 4.4.4.1b. If the average stress in the 14 tendons checked is less than 144,000 psi (60% of ultimate stress), all tendons shall be checked for stress and retensioned, if necessary, to a stress of 144,000 psi.

b. Before reseating a tendon, additional stress (6%) shall be imposed to verify the ability of the tendon to sustain the added stress applied during accident conditions.

4.4.5 Containment Isolation Valves

4.4.5.1 Each isolation valve specified in Table 3.6-1 shall be demonstrated to be operable in accordance with the Ginna Station Pump and Valve Test Program submitted in accordance with 10 CFR 50.55a.

4.4.6 Containment Isolation Response

4.4.6.1 Each containment isolation instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION, and CHANNEL FUNCTIONAL TEST operations for the MODES and at the frequencies shown in Table 4.1-1.

4.4.6.2 The RESPONSE TIME of each containment isolation function shall be demonstrated to be within the limit at least once per 18 months. Each test shall include at least one channel per function such that all channels are tested at least once every N times 18 months where N is the total number of redundant channels in a specific function as shown in the "Total No. of Channels" Column of Table 3.5-3. The response time limit shown on Table 3.6-1 does not include diesel generator starting times but does include valve travel times for all valves that change position. The times determined in independent tests, such as electronic response of portions of the initiating circuitry and valve travel times, may be combined to determine the total function response time.

4.4.7 <u>Containment Hydrogen Monitors</u>

4.4.7.1 Demonstrate that two hydrogen monitors are operable at least daily by verifying that the unit is on or in standby.

4.4.7.2 At least once per quarter perform a channel calibration using two sample gases containing known concentrations of hydrogen.

<u>Basis</u>:

The containment is designed for an accident pressure of 60 psig.[1] While the reactor is operating, the internal environment of the containment will be air at approximately atmospheric pressure. The maximum temperature of the steam-air mixture at the peak accident pressure of 60 psig is calculated to be 286°F.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 9 TO FACILITY OPERATING LICENSE NO. DPR-18

ROCHESTER GAS AND ELECTRIC CORPORATION

R. E. GINNA NUCLEAR POWER PLANT

DOCKET NO. 50-244

1.0 INTRODUCTION

In November 1980, the staff issued NUREG-0737, "Clarification of TMI Action Plan Requirements," which included all TMI Action Plan items approved by the Commission for implementation at nuclear power reactors. NUREG-0737 identifies those items for which Technical Specifications (TS) are required. A number of items which require TS were scheduled for implementation after December 31, 1981. The staff provided guidance on the scope of TS for all of these items in Generic Letter 83-37. Generic Letter 83-37 was issued to all Pressurized Water Reactor (PWR) licensees on November 1, 1983. In this Generic Letter, the staff requested licensees to:

1. review their facility's TS to determine if they were consistent with the guidance provided in the Generic Letter, and

2. submit an application for a license amendment where deviations or absence of TS were found.

By letter dated September 14, 1984 and superseded on February 21, 1985, Rochester Gas and Electric Corporation (the licensee) responded to Generic Letter 83-37 by submitting a TS change request for the Ginna plant. This evaluation covers the following TMI Action Plan items:

1. Reactor Coolant System Vents (II.B.1)
2. Noble Gas Effluent Monitors (II.F.1.1)
3. Containment High-Range Radiation Monitor (II.F.1.3)
4. Containment Pressure Monitor (II.F.1.4)
5. Containment Water Level Monitor (II.F.1.5)
6. Containment Hydrogen Monitor (II.F.1.6)
7. Instrumentation for Detection of Inadequate Core Cooling (II.F.2)
8. Control Room Habitability Requirements (III.D.3.4)

A Notice of Consideration of Issuance of Amendment to License and Proposed No Significant Hazards Consideration Determination and Opportunity for Hearing related to the requested action was published in the Federal Register on May 21, 1985 (50 FR 20987). No public comments or requests for hearing were received.

2.0 EVALUATION

1. Reactor Coolant System Vents (II.B.1)

   Our guidance for Reactor Coolant System (RCS) vents identified the need for at least one operable vent path at the reactor vessel head and the pressurizer steam space, for Westinghouse reactors. Generic Letter 83-37 also provided limiting conditions for operation and the surveillance requirements for the RCS vents. The licensee has proposed TS that are consistent with our guidance. Therefore, we find the proposed TS to be acceptable.

2. Noble Gas Effluent Monitors (II.F.1.1)

   The licensee has supplemented the existing normal range monitors to provide noble gas monitoring in accordance with TMI Action Plan Item II.F.1.1. The proposed TS for Noble Gas Effluent Monitors meet the intent of the guidelines provided in Generic Letter 83-37. Therefore, we conclude that the TS for Item II.F.1.1 are acceptable.

3. Containment High-Range Radiation Monitor (II.F.1.3)

   The licensee has installed two in-containment monitors in the Ginna plant that is consistent with the guidance of TMI Action Plan Item II.F.1.3. Generic Letter 83-37 provided guidance for limiting conditions for operation and surveillance requirements for these monitors. The licensee proposed TS that are consistent with the guidance provided in our Generic Letter 83-37. Therefore, we conclude that the proposed TS for Item II.F.1.3 are acceptable.

4. Containment Pressure Monitor (II.F.1.4)

   The Ginna plant has been provided with two wide range channels for monitoring containment pressure following an accident. The licensee has proposed TS that are consistent with the guidelines contained in Generic Letter 83-37. Therefore, we conclude that the proposed TS for containment pressure monitor are acceptable.

5. Containment Water Level Monitor (II.F.1.5)

   Narrow range and wide range containment water level monitors provide the capability required by TMI Action Plan Item II.F.1.5. The proposed TS contain limiting conditions for operation and surveillance requirements that meet the intent of the guidance contained in Generic Letter 83-37. Therefore, we conclude that the proposed TS for containment water level monitors are acceptable.

6. Containment Hydrogen Monitor (II.F.1.6)

   The licensee installed containment hydrogen monitors that provide the capability required by TMI Action Plan Item II.F.1.6. The proposed TS contain appropriate limiting conditions for operation and surveillance requirements for these monitors. We conclude that the proposed TS are acceptable as they meet the intent of the guidance contained in Generic Letter 83-37.

7. Instrumentation for Detection of Inadequate Core Cooling (II.F.2)

Generic Letter 83-37 provided the guidance on TS for the subcooling margin monitors, a reactor coolant inventory tracking system and core exist thermocouples. The licensee has submitted the TS for the subcooling margin monitors and core exist thermocouples. We have reviewed the proposed TS for the subcooling margin monitors and core exit thermocouples and conclude that the proposed TS are acceptable as they are consistent with our guidance contained in Generic Letter 83-37.

8. Control Room Habitability (III.D.3.4)

The guidance of NUREG-0737 requires assurance on the part of the licensee that control room operators will be adequately protected against the effects of an accidental release of of toxic and/or radioactive gases from sources either onsite or offsite. Generic Letter 83-37 provided guidance on the toxic gas detection system, and a control room emergency air filtration system.

The licensee has proposed TS for the chlorine, ammonia, and radioactive gas detection systems. We have reviewed the proposed TS and conclude that the proposed TS are acceptable as they meet the intent of our guidance contained in Generic Letter 83-37. The TS for control room emergency air treatment system are already included in existing TS for the Ginna Plant.

3.0 ENVIRONMENTAL CONSIDERATION

This amendment involves a change to 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 and changes to the surveillance requirements. 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, 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.

4.0 CONCLUSION

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.

5.0 ACKNOWLEDGEMENT

C. Patel and C. Miller prepared this Safety. Evaluation.

Dated: July 30, 1985