REGULATORY DOCKET FILE COPY

AUG 24 1979

Docket No. 50-244

Mr. Leon D. White, Jr.
Vice President
Electric and Steam Production
Rochester Gas and Electric Corporation
89 East Avenue
Rochester, New York 14649

Dear Mr. White:

The Commission has issued the enclosed Amendment No. 29 to Provisional Operating License No. DPR-18 for the R. E. Ginna Nuclear Power Plant. This amendment is in reference to your application dated February 1, 1977, and your submittals dated February 6, 1978 and August 25, 1978.

The amendment modifies the provisions of the Technical Specifications to incorporate the new Standby Auxiliary Feedwater System pumps, which were added as the result of our review of your analysis for high energy line breaks outside of containment. In addition to the facility modifications that were completed following analysis of the High Energy Line Break Outside Containment, you committed, by letter dated June 27, 1979, to provide jet shielding for one atmospheric steam dump valve, all steam generator code safeties, and the two main steam bypass valves and their associated 3-inch piping. This shielding would be provided in conjunction with the Systematic Evaluation Program (SEP). Furthermore, modifications to the Intermediate Building wall resulting from analysis of high energy line breaks in the Turbine Building will be made as necessary upon completion of the SEP.

We have made modifications to your application. These modifications have been discussed with representatives of Rochester Gas and Electric Corporation (RG&E) and they have agreed to the changes.

Note that the enclosed Safety Evaluation approves the structural design of the pressure shielding steel diaphragm walls and requires that you submit within 60 days a schedule for completion of these walls in the Turbine Building.

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A copy of the Notice of Issuance is also enclosed.

Sincerely,

Original Signed by:
Dennis L. Ziemann

Dennis L. Ziemann, Chief
Operating Reactors Branch #2
Division of Operating Reactors

Enclosures:
1. Amendment No. 29 to License No. DPR-18
2. Safety Evaluation (Including Appendices 1 and 2)
3. Notice of Issuance

cc w/enclosures:
See next page

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

August 24, 1979

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Vice President
Electric and Steam Production
Rochester Gas and Electric Corporation
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Operating Reactors Branch #2
Division of Operating Reactors

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2. Safety Evaluation
   (Including Appendices
   1 and 2)
3. Notice of Issuance

cc w/enclosures:
See next page

cc w/enclosures:
Lex K. Larson, Esquire
LeBoeuf, Lamb, Leiby & MacRae
1757 N Street, N. W.
Washington, D. C. 20036

Mr. Michael Slade
1250 Crown Point Drive
Webster, New York 14580

Rochester Committee for
Scientific Information
Robert E. Lee, Ph.D.
P. O. Box 5236 River Campus
Station
Rochester, New York 14627

Jeffrey Cohen
New York State Energy Office
Swan Street Building
Core 1, Second Floor
Empire State Plaza
Albany, New York 12223

*Director, Technical Development Programs
State of New York Energy Office
Agency Building 2
Empire State Plaza
Albany, New York 12223

Rochester Public Library
115 South Avenue
Rochester, New York 14604

Supervisor of the Town
of Ontario
107 Ridge Road West
Ontario, New York 14519

Director, Technical Assessment
Division
Office of Radiation Programs
(AW-459)
U. S. Environmental Protection
Agency
Crystal Mall #2
Arlington, Virginia 20460

U. S. Environmental Protection
Agency
Region II Office
ATTN: EIS COORDINATOR
26 Federal Plaza
New York, New York 10007

*W/incoming dtd. 2/1/77, 2/6/78 and 8/25/78)

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 PROVISIONAL OPERATING LICENSE

Amendment No. 29
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 February 1, 1977, 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.

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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 Provisional 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. 29, 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

Dennis L. Ziemann

Dennis L. Ziemann, Chief
Operating Reactors Branch #2
Division of Operating Reactors

Attachment:
Changes to the Technical
Specifications

Date of Issuance: August 24, 1979

ATTACHMENT TO LICENSE AMENDMENT NO. 29

PROVISIONAL OPERATING LICENSE NO. DPR-18

DOCKET NO. 50-244

Change the Technical Specifications contained in Appendix A of License No. DPR-18 as indicated below. The revised pages contain the captioned amendment number and marginal lines to reflect the area of change.

REMOVE	INSERT
3.4-1	3.4-1
3.4-2	3.4-2
3.4-3	3.4-3
--	3.4-4
4.8-1	4.8-1
4.8-2	4.8-2
--	4.8-3

3.4 **Turbine Cycle**

Applicability

Applies to the operating status of turbine cycle.

Objective

To define conditions of the turbine cycle steam-relieving capacity. Auxiliary Feedwater System and Service Water System operation is necessary to ensure the capability to remove decay heat from the core. The Standby Auxiliary Feedwater System provides additional assurance of capability to remove decay heat from the core should the Auxiliary Feedwater System be unavailable.

Specification

3.4.1 When the reactor coolant temperature is above 350°F, the following conditions shall be met:

a. A minimum turbine cycle code approved steam-relieving capability of eight (8) main steam valves available (except for testing of the main steam safety valves).

b. Three auxiliary feedwater pumps and their associated flow paths (including backup supply from the Service Water System) must be operable.

c. A minimum of 15,000 gallons of water shall be available in the condensate storage tanks for the Auxiliary Feedwater System.

d. Two Standby Auxiliary Feedwater pumps and associated flow path (including flow path from the Service Water System) must be operable.

3.4.2 Actions To Be Taken If Conditions of 3.4.1 Are Not Met

a. With one or more main steam code safety valves inoperable, restore the inoperable valve(s) to operable status within 4 hours or be in at least hot shutdown within the next 6 hours and in cold shutdown within the following 30 hours.

b. With one auxiliary feed pump inoperable, restore the pump to operable status within 7 days. If the pump is not restored to operable status within 7 days submit a Thirty Day Written Report in accordance with Specification 6.9.2 outlining the cause of the inoperability and plans for restoring the pump to operable status.

c. With two auxiliary feed pumps inoperable, restore two pumps to operable status within 72 hours or be in hot shutdown within the next 12 hours (and in cold shutdown within the following 24 hours).

d. With one standby auxiliary feed pump inoperable, restore two pumps to operable status within 7 days or be in hot shutdown within the next 12 hours and cold shutdown within the following 24 hours.

e. With the required 15,000 gallons of water unavailable to condensate storage tanks, within 4 hours, either:

   1. Restore the required amount of water or be in hot shutdown within 12 hours, or

   2. Demonstrate the operability of the Service Water System as a backup supply to the auxiliary feed system and restore the required amount of water in the condensate storage tanks within 7 days or be in hot shutdown within the following 12 hours.

Basis:

A reactor shutdown from power requires removal of core decay heat. Immediate decay heat removal requirements are normally satisfied by the steam bypass to the condenser. Therefore, core decay heat can be continuously dissipated via the steam bypass to the condenser as feedwater in the steam generator is converted to steam by heat absorption. Normally, the capability to return feedwater flow to the steam generators is provided by operation of the turbine cycle feedwater system.

The eight main steam safety valves have a total combined rated capability of 6,580,000 lbs/hr. This capability exceeds the total full power steam flow of 6,577,279 lbs/hr. In the event of complete loss of off-site electrical power to the station, decay heat removal is assured by either the steam-driven auxiliary feedwater pump or one of the two motor-driven auxiliary feedwater pumps, and steam discharge to the atmosphere via the main steam safety valves or atmospheric relief valves.(1)(2) The turbine driven pump can supply 200% of the required feedwater and one motor-driven auxiliary feedwater pump can supply 100% of the required feedwater for removal of decay heat from the plant, so any combination of two pumps can remove decay heat with a postulated single failure of one pump. The minimum amount of water in the condensate storage tanks is the amount needed to remove core residual and decay heat for 1/2 hour after reactor scram from full power. If the outage is more than 1/2 hour, Lake Ontario water will be used. An unlimited supply is available from the lake via either leg of the plant service water system for an indefinite time period.

The Standby Auxiliary Feedwater System is provided to give additional assurance of the capability to remove decay heat from the reactor. The system would be used only if none of the auxiliary feedwater pumps were available to perform their intended function. Since operability requirements are established for the auxiliary feedwater system, the Standby System would be required only if some unlikely event should disable all auxiliary feedwater pumps. The specified time to restore the Standby System to full capability is longer than for other components since the probability of being required to use the Standby System is extremely low.[3]

References:

(1) FSAR Section 10.4

(2) FSAR Section 14.1.9

(3) "Effects of High Energy Pipe Breaks Outside the Containment Building" submitted by letter dated November 1, 1973 from K. W. Amish, Rochester Gas and Electric Corporation to A. Giambusso, Deputy Director for Reactor Projects, U. S. Atomic Energy Commission

4.8 **Auxiliary Feedwater Systems**

Applicability

Applies to periodic testing requirements of the turbine-driven and motor-driven auxiliary feedwater pumps and of the Standby Auxiliary Feedwater pumps.

Objective

To verify the operability of the auxiliary feedwater system and the Standby Auxiliary Feedwater System and their ability to respond properly when required.

Specification

4.8.1. Except during cold or refueling shutdowns each motor driven auxiliary feedwater pump unless it is declared inoperable without testing will be started at intervals not to exceed one month and a flow rate of 200 gpm established.

4.8.2 Except during cold or refueling shutdowns the steam turbine driven auxiliary feedwater pump unless it is declared inoperable without testing will be started at intervals not to exceed one month and a flow rate of 400 gpm established.

4.8.3 Except during cold or refueling shutdowns the auxiliary feedwater pumps suction discharge and crossover motor operated valves shall be exercised at intervals not to exceed one month.

4.8.4 Except during cold or refueling shutdowns each Standby Auxiliary Feedwater pump unless it is declared inoperable without testing, will be started at intervals not to exceed one month and a flow rate of 200 gpm established.

4.8.5 Except during cold or refueling shutdowns, the suction, discharge, and cross-over motor operated valves for the Standby Auxiliary Feedwater pumps shall be exercised at intervals not to exceed one month.

4.8.6 These tests shall be considered satisfactory if control board indication and subsequent visual observation of the equipment demonstrate that all components have operated properly. These tests shall be performed prior to exceeding 5% power during a startup if the time since the last test exceeds one month.

4.8.7 At least once per 18 months, control of the standby auxiliary feed system pumps and valves from the control room will be demonstrated.

Basis

The monthly testing of the auxiliary feedwater pumps by supplying feedwater to the steam generators will verify their ability to meet design. The flow rates will be measured at a simulated steam generator pressure of 1100 psia. The capacity of any one of the three auxiliary feedwater pumps is sufficient to meet decay heat removal requirements. Proper functioning of the steam turbine admission valve and the feedwater pumps start will demonstrate the integrity of the steam drive pump.

Monthly testing of the Standby Auxiliary Feedwater pumps by supplying water from a condensate supply tank to the steam generators will verify their ability to meet design. The flow rate will be measured at a simulated steam generator pressure of 1100 psia. The Standby Auxiliary Feedwater pumps would be used only if all three auxiliary feedwater pumps were unavailable. One of the two Standby pumps would be sufficient to meet decay heat removal requirements. Proper functioning of the suction valves from the service water system, the discharge valves, and the crossover valves will demonstrate their operability.

Verification of correct operation will be made both from instrumentation within the main control room and by direct visual observation of the pumps.

<u>References</u>:

FSAR - Section 10.4

FSAR - Section 14.1.9

FSAR - Section 14.2.5

"Effects of High Energy Pipe Breaks Outside the Containment Building" submitted by letter dated November 1, 1973 from K. W. Amish, Rochester Gas and Electric Corporation to A. Giambusso, Deputy Director for Reactor Projects, U. S. Atomic Energy Commission.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 29 TO PROVISIONAL 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 18, 1972, the Atomic Energy Commission's* Regulatory staff requested Rochester Gas and Electric Corporation (RG&E)(1) (licensee) to submit a detailed evaluation to substantiate that the R. E. Ginna Nuclear Power Plant (Ginna) could withstand the effects of a postulated rupture of any high energy fluid piping outside the primary containment, including the double ended rupture of the largest line in the main steam and feedwater systems. It was further requested that, if the results of the evaluation indicated changes to the facility were necessary to assure safe plant shutdown, information on the design changes and plant modifications be provided. Criteria for performing this evaluation were included in our December 18, 1972 letter. NRC and RG&E representatives met in Bethesda, Maryland, on February 1, July 18 and September 18, 1973, to discuss the NRC request and the scope of the expected RG&E analyses.

In response to our request, RG&E submitted a letter(2) dated November 1, 1973, that included a summary report "Effects of Postulated Pipe Breaks Outside the Containment Building" dated October 29, 1973. The results of this pipe whip and building pressurization analysis indicated that the intermediate building structure at Ginna was generally incapable of resisting pipe whip and pressurization effects of most postulated main steam and feedwater breaks within this building and from the adjacent turbine room. The licensee determined that modification of the structure or pipe encapsulation to provide the required protection was not practical(3) and an extensive volumetric examination program** to provide added assurance that the postulated piping system breaks would not occur was later proposed(4), initiated in 1973 and finally approved(5) by NRC in 1975.

*Currently known as the Nuclear Regulatory Commission (NRC).
**In accordance with the requirements of 10 CFR Part 50, Section 50.55a, paragraph (g), RG&E submitted by letter dated 7/2/79, the "Ginna Station In-Service Program for the 1980 through 1989 Interval".

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Additional information[6] was submitted by RG&E's letter dated May 24, 1974. This information was responsive to NRC concerns for postulated high energy line breaks outside containment and potential effects on safety related equipment that might be required to cool the core. The licensee later submitted a schedule[7] for analysis and plant modifications. As a result of the High Energy Line Break Outside of Containment evaluation, plant changes have been made[19] as summarized below:

-- An augmented In-Service-Inspection Program has been initiated[4] to further reduce the probability of a main feedwater or steam line rupture.

-- A Standby Auxiliary Feedwater System has been added to further improve steam generator feedwater reliability and specifically to substitute for the auxiliary feedwater in the low probability that auxiliary feedwater pumps are damaged due to nearby high energy pipe breaks within the intermediate building.

-- Check valves have been added to existing auxiliary feedwater lines near the connections to the main feedwater lines to minimize the auxiliary feedwater piping that is pressurized during normal operation.

-- Two parallel remotely operated valves have been added to a crossover line between the motor driven pump discharges to provide additional auxiliary feedwater makeup capability.

-- A large metal plate jet shield has been installed underneath the main steam header in the Intermediate Building to protect the service water piping from a postulated crack in the main steam line. Jet Impingement Shields have been added to protect vital equipment including containment isolation valves, motor generators, transfer switches, cable trays, terminal boxes and wiring, pressure transmitters and reactor trip breakers. Also jet shields have been added to protect main steam bypass valves and piping and other locations listed by RG&E.

-- Instrument cabling has been relocated to areas that will not be affected by postulated high energy pipe breaks.

-- The heating and ventilation system has been modified to withstand postulated high energy pipe breaks without further endangering the capability to safely shut down the plant.

-- The east end of the cable tray that connects the Intermediate Building and the Relay Room of the Control Building has been sealed to prevent damage that could result from a postulated high energy line break.

-- Openings around pipes and cable trays that pass through the areas required for safe shutdown of the plant have been sealed to prevent steam leakage into these areas in the unlikely event of steam or feedwater line breaks in the Turbine Building.

-- Steam generator blowdown lines have been rerouted through the sub-basement to minimize the potentially detrimental effects of breaks in these lines within the Intermediate Building.

-- Sufficient floor grating has been installed at manholes to guard against flooding of safety related equipment in the Intermediate Building resulting from an assumed feedwater line break.

-- Steam line pressure and feedwater flow transmitters have been relocated away from the locations that could be affected by postulated high energy line breaks.

-- Pressure shielding steel diaphragm walls are being installed at selected locations in the Turbine Building to assure continued operability of safety related equipment following a postulated high energy pipe break in the Turbine Building.

-- RG&E committed, by letter dated June 27, 1979, to provide jet shielding for one atmospheric steam dump valve, all steam generator code safeties, and the two main steam bypass valves and their associated 3-inch piping. This shielding would be provided in conjunction with the Systematic Evaluation Program (SEP). Furthermore, modifications to the Intermediate Building wall resulting from analysis of high energy line breaks in the Turbine Building will be made as necessary upon completion of the SEP.

Discussion

Ginna is a pressurized water reactor that utilizes a reinforced concrete containment which contains the entire primary coolant system, including the steam generators.

The criteria and requirements used by the licensee and the staff for evaluating the high energy line breask outside containment are summarized as follows:[5]

1. Equipment and structures necessary to shut down the reactor and maintain it in a safe shutdown condition, assuming a concurrent and unrelated single active failure of essential equipment, should

be protected from all effects of ruptures in pipes carrying high energy fluid, up to and including a double-ended rupture of such pipes, where the service temperature and service pressure conditions of the fluid exceed 200°F and 275 psig. Breaks should be assumed to occur in those locations specified in the "pipe whipe criteria". The rupture effects to be considered include pipe whip, structural (including the effects of jet impingement), and environmental.

2. In addition, equipment and structures necessary to shut down the reactor and maintain it in a safe shutdown condition, assuming a concurrent and unrelated single active failure of essential equipment, should be protected from the environmental and structural effects (including the effects of jet impingement) resulting from a single open crack at the most adverse location in pipes normally carrying high energy fluid routed in the vicinity of this equipment. The postulated size of the cracks was either 1/2 the pipe diameter in length and 1/2 the wall thickness in width (critical crack size) or equivalent pipe flow cross section in area.

The licensee evaluated all piping outside containment that contains high energy fluid and is in the same building with or in the proximity of safety related equipment required for safe shutdown. These lines are:

Main Steam
Feedwater
Auxiliary Feedwater
Steam Supply to Auxiliary Feedwater Pump Turbine
Steam Generator Blowdown
Charging Line
Plant Steam

The licensee's evaluation postulated longitudinal and circumferential breaks at high stress locations specified by the NRC criteria for piping break locations and considered the effects of pipe whip, jet impingement, pressurization, environment and flooding. For the evaluation of piping cracks, effects of pipe whip and pressurization were not applicable. The licensee described the course of events following various size breaks of the main steam and feedwater lines at different reactor operating conditions. The equipment necessary to bring the plant to a safe shutdown was listed. The licensee's analyses indicate that the Intermediate Building, through which the main steam and feedwater lines pass from the Containment Building to the Turbine Building, cannot withstand most main steam line and feedwater line breaks. This results from the pressurization of the building following the postulated high energy pipe break exceeding the design pressure for the concrete block walls and the roof, and from the structural capabilities not being sufficient to withstand the effects of pipe whip. Some equipment that is used to maintain the reactor in a safe shutdown condition is located in this building and might be rendered inoperable. This equipment includes instrument channel cables, service water piping, and the auxiliary feedwater system. A number of alternatives to the final plant modifications[2] were evaluated and considered by the licensee to be impractical[3].

Evaluation

The Augmented In-Service Inspection Program[4] proposed and implemented by the licensee consists of radiographic examination of all welds at the design basis break locations in the main steam and feedwater lines and at other locations where a failure would result in unacceptable consequences. The examination techniques, procedures, and inspection intervals are based on the requirements of Class 2 components of Section XI of the ASME Boiler and Pressure Vessel Code. The program* is based on ten year inspection intervals with the first interval running from 1973 to 1982. The extensive in-service inspection program is designed to preclude design bases or consequential main steam or feedwater pipe breaks.

During each third of the first inspection interval, the program provides for examination of all welds at specified design basis break locations and one-third of all the welds at specified locations where a weld failure could result in unacceptable consequences. During each one-third of the succeeding 10-year intervals, the program provides for examination of one-third of the welds at design basis break locations but continues unchanged with one-third of the welds at locations where a weld failure could result in unacceptable consequences. This program is designed to detect flaws capable of causing pipe failure. The frequency of reinspections is designed to detect any change in condition in advance of a potential failure. We have concluded that this augmented inspection program is a prudent measure to ensure a very low probability of any break in the main steam and feedwater lines. The inspection requirements for this program have been incorporated into the Technical Specifications[5].

The Instrumentation Channels that initiate the protective action in the event of a main steam line or feedwater line break are: Pressurizer Pressure, Steam Line Pressure, Steam Line Flow, Feedwater Flow, Pressurizer Water Level, and Steam Generator Water Level.

The pressurizer pressure, steam line flow, pressurizer water level and steam generator water level transmitters are located inside containment and, therefore, their operability would not be affected by a high energy line break outside the containment. Some of the signal cables from these transmitters, however, are routed through cable trays in the Intermediate Building. To ensure that the minimum number of these channels required to produce the protective actions (safety injection, reactor trip, and feedwater and steam line isolation) are not adversely affected by a high energy line break in the Intermediate Building, their signals have been rerouted out of other containment penetrations and do not pass through the Intermediate Building.

*RG&E letter dated 7/2/79 presents the 1980-1989 Ginna ISI Program.

The Steam Line Pressure and Feedwater Flow Transmitter Signal Cables have been relocated to areas with no high energy lines. The sensing lines for the transmitters are susceptible to damage since they connect to high energy lines. However if they rupture, the channels fail downscale and since low steam line pressure and low feedwater flow produce the trips for protective action, the channels fail in the safe direction. In addition, the signal cables for a cold leg reactor coolant temperature channel from each loop have been rerouted outside the Intermediate Building to provide the operators with additional information to follow the course of the accident.

The following instrument channels are isolated from the effects of high energy line break outside containment.

Instrument	No. of Protected Channels	No. Required to Trip
Steam Generator Level	2 per loop	2 per loop 1 per loop with Steam Flow-Feed Flow Mismatch
Steam Line Flow	2 per loop	1 per loop
Feedwater Flow	2 per loop	1 per loop
Steam Line Pressurizer	3 per loop	2 per loop
Pressurizer Pressure	2 per loop	1 per loop
Pressurizer Level	1 per loop	1 per loop
Reactor Coolant Temperature	2 per loop	NA

The instrument channels or signal cables that remain in the unprotected areas of the Intermediate Building are likely to perform their trip function by providing protective action signals for the steam or feedwater line breaks either in the normal fashion or by the fail-safe trip. This is because any failure which could occur would most likely be a separation of the sensing line or signal cable and, except for the steam flow channels, loss of signal trips the channel. Also the required protective actions can be initiated by the response of a single one of the parameters monitored by the channels above, such as low steam pressure on two channels in one loop, or by a number of diverse responses, such as low pressurizer pressure and level on one channel. Therefore, the protected channels and those remaining in the unprotected area maintain the required diversity and redundancy for reactor protection systems. In addition, the protected channels will ensure that the operator is provided with information for the course of the accident. On this basis, we find these modifications acceptable.

The Auxiliary Feedwater System is also located in the Intermediate Building with all three pumps in the same vicinity. There are two motor driven pumps and one steam driven pump. These pumps are only used during start-up and normal or emergency shutdown of the plant. The pumps are susceptible to damage from the effects of breaks in the main steam and feedwater lines and the auxiliary steam and feedwater lines. To ensure the heat removal capability for core cooling, the licensee proposed and later installed a Standby Auxiliary Feedwater System adjacent to the Auxiliary Building along the south wall. The Standby Auxiliary Feedwater Pumphouse is a seismic Class I concrete structure supported by caissons(9).

The Standby Auxiliary Feedwater System consists of two, independent 100 percent capacity subsystems in a new structure remote from high energy lines. The discharge piping from the pumps was routed through the Auxiliary Building, enters the containment through penetrations remote from the main steam and feedwater lines, and connects to the feedwater lines near each steam generator with check valves near the connection to minimize the amount of line pressurized during normal plant operation.

The pumps take suction from the Service Water Loops inside the Auxiliary Building are motor driven from the Engineered Safety Features busses, and are manually started from the control room in the event that the Auxiliary Feedwater pumps, which start automatically, are not operable. The analysis performed by the licensee assumes that feedwater is not available for 10 minutes following the worst case line break. This is ample time for the control room operator to take action since alarms and indications are available in the control room to alert the operator to the lack of effective auxiliary feedwater flow and the standby pumps can be put into operation from the control room.

Our concerns for the structural, mechanical and material aspects of the modifications were adequately addressed(10) by the RG&E letter dated July 28, 1978, in response to our request dated June 21, 1978.

In the event of loss of off-site power, the pumps would be powered by the diesel generators. The diesel generators have sufficient capacity for this additional 225 Kw load. However, to prevent an overload of the feedbreakers tying the diesels to the buses, an interlock has been installed to prevent starting a standby pump when its associated auxiliary pump is running on the diesel.

The Standby Auxiliary Feedwater Pump Building and System design satisfied the codes and standards applicable in 1974 when the building was designed(9,10). We conclude that these modifications provide an acceptable backup to the Auxiliary Feedwater System for maintaining the plant in a safe shutdown condition. The scope of the Safety Evaluation of the Standby Auxiliary Feedwater System is presented in the enclosed Appendix 1. On the basis of this evaluation, the Technical Specification changes proposed by RG&E(8), which we revised with RG&E concurrence, are acceptable. Also, the same operating procedure requirements for the prevention of water hammer in the Auxiliary Feedwater System should be applied to the Standby Auxiliary Feedwater System.

The Ventilation Systems were evaluated to determine whether the steam from high energy line breaks would intrude into an area where personnel or equipment important to safety would be endangered. It was determined that modifications were necessary to the control room lavatory exhaust, the control building ventilation equipment room relief opening, the relay room cable tray openings and tunnel, the battery room exhaust and cable tray openings, the diesel generator room piping and cable tray openings, and some interconnecting ventilation ducts between the Intermediate Building and the Auxiliary Building. All of these openings have been sealed and the exhausts have been ducted to areas not subject to intrusion of the steam from a high energy line break. Based on the above, we conclude that these modifications reduce the probability of adverse consequences from the postuated high energy line breaks and are, therefore, acceptable.

Pressure Shielding Steel Diaphragm Walls were proposed by RG&E's letter dated February 6, 1978[11]. The steel diaphragm walls were to have been erected between the Control Building and the Turbine Building and between the Diesel Generator Rooms and the Turbine Building. The walls would:

-- Comply with the requirements for physical protection of licensed activities against industrial sabotage (10 CFR Section 73.55)

-- Provide protection from postulated fires on the operating level of the Turbine Building

-- Provide protection from postulated high energy line breaks in the Turbine Building

We met with representatives of the licensee in Bethesda, Maryland, on February 15, 1978, to discuss fire protection and structural aspects of the diaphragm wall and on January 30, 1979, to discuss structural design criteria for the wall. On the basis of information provided by the licensee[12, 14] we have concluded[15] that the steel wall and door that have recently been added between the Control Room and the Turbine Building are designed for high power rifle resistance (level IV bullet resistance) and, therefore, meet the requirements of 10 CFR 73.55.

We have reviewed the adequacy of the Steel Diaphragm walls between the Control Building and the Turbine Building and between the Diesel Generator Rooms and the Turbine Building with respect to fire protection. Based on the information provided by the licensee[11, 12, 13, 14], we have concluded[16] that the concept of a steel diaphragm wall between the Turbine Building and the Control Room protected by an automatically actuated water curtain is acceptable, but the details of the water supply and actuation system must be submitted for our review. Concerning the Pressure Shielding Steel Diaphragm Turbine Building walls adjacent to the Diesel Generator, Relay and Battery Rooms, the licensee has agreed[16] to conduct studies to determine what active and passive systems should be installed to prevent structural failure from fire that would jeopardize safe shutdown of the plant. We have also identified[16] the requirements for fire doors in the areas where the steel diaphragm wall is being constructed.

The NRC Safety Evaluation of the structural adequacy of the Pressure Shielding Steel Diaphragm Walls is presented in the enclosed Appendix 2. We have concluded on the basis of information presented in licensee letters[11, 12, 13, 14] and during a meeting with NRC representatives[17], that the structural criteria and design methods for the steel diaphragm walls are adequate to assure safe shut down of the reactor following a high energy pipe break in the Turbine Building. However, our conclusion is based on the premise that the peak Turbine Building pressure and temperatures that the Turbine Building steel diaphragm walls adjacent to the Diesel Generator Rooms and the Control Building (Control Room, Relay Room and Battery Room) must withstand, results from a postulated rupture of the 20" Feedwater Line. Since the licensee had previously reported[6] that the pressure on the operating level of the Turbine Building as a result of a break in the 24" or 36" steam line peaked at 0.098 psig with steam relief through the building exhaust fans in the wall and roof, and later reported 0.70 psig pressure peaks[17], resulting from a break in the 20" main feedwater line, we requested RG&E to submit additional analysis. The licensee's basis[10] for using the Main Feedwater 20" pipe break to determine peak Turbine Building transient pressure and temperature for the structural design of the new steel diaphragm walls was justified because of the augmented In-Service Inspection of all welds in the steam lines in the Turbine Building and the resultant low probability of a large break in the steam lines. Nevertheless, at our request, by letters dated May 17, 1979 and July 6, 1979, the licensee provided supplementary information[18,20] which in addition to the Turbine Building pressure transient analyses for postulated feedwater pipe breaks, also included steam line breaks in the Turbine Building. As expected, these calculations showed that the steam line break pressure transients were significantly greater than originally reported[6].

The following additional information[17, 18, 20] provided by RG&E:

-- The peak pressure transients in the Turbine Building calculated by the licensee are less than the 0.7 psig structural design pressure for the steel diaphragm wall on the mezanine floor along the control room wall and less than the 1.14 psig structural design pressure for the steel diaphragm walls on the operating floors at the relay, battery and diesel generator room walls.

-- The new steel diaphragm walls are at nearly opposite ends of the Turbine Building from the high energy piping thereby providing adequate separation to preclude wall damage at these locations because of pipe whip or jet impingement that could accompany a high energy pipe break in the Turbine Building.

Based on this information and our detailed Safety Evaluation of the pressure shielding steel diaphragm walls in the Turbine Building which is included as Appendix 2 to this Safety Evaluation, we have concluded that the structural adequacy of the steel diaphragm walls as described by the licensee[10] is acceptable. A schedule for completion of the installation of the steel diaphragm walls, in accordance with the information provided, should be submitted within 60 days.

Environmental Consideration

We have determined that the amendment does not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that the amendment involves an action which is insignificant from the standpoint of environmental impact, and pursuant to 10 CFR §51.5(d)(4), that an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of this amendment.

Conclusion

We have concluded, based on the considerations discussed above, that: (1) because the amendment does not involve a significant increase in the probability or consequences of accidents previously considered and does not involve a significant decrease in a safety margin, the amendment does not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) 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.

Enclosures:
1. Appendix 1, "Detailed Evaluation of the Standby Auxiliary Feedwater System - R. E. Ginna"
2. Appendix 2, "Detailed Evaluation of the Pressure Shielding Steel Diaphragm in Turbine Building - R. E. Ginna"

Date: August 24, 1979

REFERENCES:

1. NRC letter to RG&E, dated December 18, 1972, "Effects on Essential Auxiliary Systems of a Major Break of the Largest Steam or Feedwater Line".

2. RG&E letter to NRC dated November 1, 1973, "An Analysis of the Effects of Postulated Pipe Breaks Outside Containment".

3. RG&E letter to NRC dated September 4, 1974, "High Energy Line Break Outside Containment - Alternate Possibilities for Protection from High Energy Line Breaks".

4. RG&E letter to NRC dated October 31, 1974, "Proposed In-Service Inspection Program for High Energy Piping Outside Containment".

5. NRC Amendment No. 7 to Provisional Operating License No. DPR-18 for the R. E. Ginna Nuclear Power Plant, dated May 14, 1975.

6. RG&E letter to NRC dated May 24, 1974 - RG&E response to NRC staff request for additional information (April 24, 1974) related to "Effects of Postulated Pipe Breaks Outside Containment".

7. RG&E letter to NRC dated November 1, 1974, "Updated Schedule for Analysis and Plant Modifications".

8. RG&E letter to NRC dated February 1, 1977, the third of three RG&E requests "Technical Specification Changes to Include Requirements for the Standby Auxiliary Feedwater System" where it is declared operational.

9. RG&E letter to NRC dated May 20, 1977, "Standby Auxiliary Feedwater System Design Criteria".

10. RG&E letter to NRC dated July 28, 1978, "Additional Information Related to Strcutural, Mechanical and Material Aspects of the Standby Auxiliary Feedwater System Modification Responsive to NRC Request dated June 21, 1978".

11. RG&E letter to NRC dated February 6, 1978, "Pressure Shielding Steel Diaphragm in Turbine Building R. E. Ginna Nuclear Power Plant, Unit No. 1, Docket No. 50-244 - Design Criteria, Rev. 2 July 18, 1977".

12. RG&E letter to NRC dated August 25, 1978, "Request for NRC Approval of the Proposed Turbine Building Modifications in the Vicinity of the Control Building and the Diesel Generator Annex.

REFERENCES:

13. RG&E letter to NRC dated October 11, 1978, "Responses to NRC Letter dated September 13, 1978, Concerning the Pressure Shielding Steel Diaphragms.

14. RG&E letter to NRC dated October 18, 1978, "Request for NRC Approval of the Fire Protection Aspects of the Control Room and Relay Room Doors that Provide Pressurization and Fire Protection.

15. NRC Amendment No. 25 to Provisional Operating License No. DPR-18 for the R. E. Ginna Nuclear Power Plant dated February 22, 1979 - Physical Security Plan.

16. NRC Amendment No. 24 to Provisional Operating License No. DPR-18 for the R. E. Ginna Nuclear Power Plant dated February 14, 1979 - Fire Protection.

17. NRC "Summary of Meeting Held on January 30, 1979" dated February 9, 1979.

18. RG&E letter to NRC dated May 17, 1978, "Calculated Pressure Transients in Turbine Building" responsive to NRC commitment (Ref. 17).

19. RG&E letter to NRC dated June 27, 1979, "Plant Modifications Resulting from High Energy Line Break Outside Containment Evaluation".

20. RG&E letter to NRC dated July 6, 1979, "Pressure Transients in the Turbine Building - Pressure Shielding Steel Diaphragms".

APPENDIX 1

DETAILED EVALUATION OF THE STANDBY AUXILIARY FEEDWATER SYSTEM
R. E. GINNA

Background

Rochester Gas & Electric (RG&E, the licensee) in its report "Effects of Postulated Pipe Breaks Outside the Containment Building for the Robert E. Ginna Nuclear Power Plant Unit No. 1", dated October 29, 1973 (Reference 1), determined that the Auxiliary Feedwater System (AFS) in the intermediate building could be damaged by a high energy line break (HELB) in that building. Therefore, RG&E has installed a standby Auxiliary Feedwater System (SAFS), independent of and remotely located from the AFS. The SAFS, described in References 1 through 4, was installed solely as a backup to the AFS in the event of a HELB and is not intended to be used to mitigate the effects of other plant accidents or transients.

The SAFS was designed and constructred for the purpose of providing feedwater to the steam generator in the event that a large main steam or feedwater line break in the Intermediate Building were to disable both trains of the AFS. The licensee has designed the SAFS as a safety related system and the system and its enclsoing structure were built to the most current criteria at the time. Under the Systematic Evaluation Program, all safety related equipment and structures at Ginna are being re-evaluated (Reference 5). Because of this ongoing re-evaluation, the SAFS and its enclosing structure were examined (1) to determine if SAFs operation would acceptably mitigate the consequences of AFS damage which could result from a postulated HELB, (2) to determine that the addition of the SAFS and its enclosing structure would not adversely impact other, previously approved, safety related systems, structures or components, and (3) to determine if the technical specification requirements for SAFS operability and surveillance were acceptable.

Description and Evaluation of Standby Auxiliary Feedwater System

The SAFS provides two independent feedwater flow paths from separate seismic Class I Service Water System loops, via two motor driven pumps to the Ginna plant's two steam generators. Each SAFS pump can, within a few minutes of reactor trip, provide the required flow for removing reactor core decay heat. In addition, a comparison of the SAFS with the AFS, which has almost ten years of operating experience, shows that each of the SAFS pumps has sufficient capacity to cool the plant adequately. Each pump supplies flow through its normally open discharge valve, a containment isolation stop check valve, and a check valve to the main feed line for its respective steam generator. Also, two motor operated cross-connect valves in parallel can be used to direct flow from one pump to either steam generator. Two manually operated valves in series can be opened to cross-connect SAFS pump suction lines. The system layout and capacity provide adequate redundancy to accommodate a single active component failure without loss of system function.

A condensate storage tank is used to store 10,000 gallons of condensate quality water for periodic SAFS testing. The tank is located in the Auxiliary Building Addition. Piping is provided to supply 125 psig condensate to the SAFS to pressurize the pump suction lines and to fill the condensate storage tank.

The SAFS piping is of ASME Code, Section III, Safety Class 2 and 3 design except for two interfaces with non-nuclear safety class piping. These interfaces are used for system pressurization from the condensate system and for pump flow recirculation testing, which are non-essential system functions. These non-essential portions of the system can be isolated to permit continued system function in the event of their failure. The SAFS instrumentation is capable of detecting significant leakage from the system, and system leakage is directed to the waste holdup tank in the Auxiliary Building basement.

The SAFS system satisfies the requirements for quality group and seismic classification as identified in Regulatory Guides 1.26 and 1.29. The seismic classification is based on the definitions provided in the facility's Final Design and Safety Analysis Report (Reference 6).

The SAFS was designed for installation with no degradation in the design or function of existing systems and for operation within code allowable stresses over the full range of expected operating temperatures. Welding procedures used to fabricate the system were in accordance with Section III of the ASME Code. Pre-operational hydrostatic testing of the ASME III Class 2 and 3 portions of the system was in accordance with the ASME Code. The SAFS discharge lines were routed through the primary containment boundary via existing spare penetrations. Local leak testing of these penetrations is conducted in accordance with 10 CFR 50, Appendix J. Material selection for the Class 2 and 3 components of the SAFS was in accordance with Section III of the ASME Code.

The licensee has not evaluated the SAFS itself with regard to the effects of pipe whip and jet impingement except for the piping inside containment between the main feedwater line and the closest check valve in the SAFS injection lines. This is in accordance with the provisions of current NRC criteria (Reference 7) for systems which do not operate during normal plant conditions. The section of SAFS piping inside containment which is a high energy line during normal plant conditions has been evaluated by the licensee and found acceptable from the standpoint of HELBs because of its location away from safe shutdown equipment. The SAFS itself is protected from the effects of HELBs outside containment because there are no high energy lines in the vicinity of SAFS components.

The design of the SAFS does not preclude feed system waterhammer. The occurrence of waterhammer is currently prevented by AFS operating procedures. Similar procedures (requiring an upper limit on feedwater addition rate whenever steam generator level is below the feed ring) would be used for the SAFS. The licensee has completed (February 1979 outage) steam generator modifications to further limit the potential for occurrence of feed system waterhammer (Reference 9), in accordance with current requirements for operating reactor plants.

The electrical power for each SAFS train is supplied by one of the two redundant 480 VAC emergency power systems at Ginna. SAFS pumps C and D are powered from existing spare feeders on emergency busses 14 and 16 respectively. These busses can receive power from offsite or onsite (diesel) sources. To prevent exceeding electrical load limits on the feeder breakers tying the diesel generators to the buses electrical interlocks are provided to prevent an AFS and a SAFS pump from being connected to the same diesel generator simultaneously. The requirement for power supply diversity (Reference 8) is not applicable to the SAFS because the requirement is based on an assumed loss of all AC power (offsite and onsite). If this is assumed, the SAFS is rendered inoperable, and the turbine driven pump of the AFS provides the required power supply diversity. The staff considers the simultaneous occurrence of a HELB which renders the AFS turbine driven pump inoperable and the total loss of both onsite and offsite AC power to be sufficiently unlikely as not to be credible.

The SAFS is manually started from the control room or from a local station in the Auxiliary Building Addition pump room. A switch for transfer of control from local to the control room is provided at the local control station. Control room indication shows the status of this transfer switch. A TEST/NORMAL mode switch is also provided local to the pumps. The switch will be in the NORMAL position at all times except when system operational tests are being conducted. In the TEST mode, interlocks will prevent pump operation unless the corresponding manual suction valve in the tank outlet line is in the fully open position, and will trip the pumps when tank LO level is reached. In the NORMAL mode, interlocks will prevent pump operation unless the motor operated valve in the corresponding service water line is open, and will also prevent two auxiliary feedwater pumps from being connected to the same diesel generator simultaneously.

Control room instrumentation will alert the operator that the AFS is ineffective and the SAFS should be started.

Pump discharge flow and pressure indication is provided locally and in the main control room. In the event of damage to a steam generator or associated piping the operator will use this flow indication together with steam generator level and steam pressure to determine which steam generator system is damaged and isolate the feedwater flow to that steam generator, using the motor operated valves if necessary. There is no provision for either the AFS or the SAFS to automatically terminate flow to a depressurized steam generator and automatically provide flow to the intact steam generator. This is accomplished by the control room operator. The effect of the lack of automatic switching of flow to the intact steam generator will be assessed in the SEP main steam line break evaluation for Ginna.

The instrumentation and controls for the SAFS conform to General Design Criteria (GDC) 19, "Control Room". The SAFS is a manually initiated system intended to be used in the event that a postulated HELB in the Intermediate Building were to disable the AFS. An analysis of a worst case feedwater line break using conservative assumptions was provided by the licensee in Reference 2. This analysis has shown that a 10 minute delay in the initiation of auxiliary feed flow from one motor driven pump results in acceptable consequences. The loss of feed accident analysis presented in Chapter 14 of Reference 6 assumes automatic initiation of one motor driven auxiliary feed pump one minute after accident initiation and results in no loss of decay heat removal capability of the steam generator receiving auxiliary feed and no loss of coolant from the primary system pressurizer relief valves. The latter analysis covers a spectrum of loss of feed events from those of high probability of occurrencee to low probability events including pump failures, valve malfunctions, loss of offsite power and pipe breaks; and, for these, the automatically initiated AFS provides adequate protection. The SAFS has been installed to protect against the low probability event of a postulated main feed or steam line break in the intermediate building that completely disables the AFS. We have reviewed the sequence of actions that the control room operator must take to initiate the SAFS and concluded that the operator would have sufficient time for manual initiation of the system within the conservatively calculated 10 minute period. Therefore, manual initiation of the SAFS is acceptable. Since the acceptance criteria for feedwater line break analyses have changed since the 1974 analysis submitted by the licensee in support of the 10 minute period for SAFS initiation, we have also reevaluated the consequences of this accident assuming core damage. Using the assumptions of instantaneous release of 2% of the iodines and noble gases in the core, a reactor coolant volume of 5750 cubic feet, primary to secondary leak rate at the maximum technical specification limit, and a relative concentration X/Q due to atmospheric diffusion of 1×10^{-3} sec/m^3, we calculate the resulting doses at the nearest site boundary to be within 10 CFR Part 100 criteria for offsite radiation doses.

Even though two actions are required to initiate the SAFS (open suction valve, start pump), the system conforms to Regulatory Guide 1.62. This is because system initiation depends on operation of a reasonable minimum of equipment considering the desire to avoid introduction into the steam generator of chemical impurities from the Service Water System through a single operator error or a single electrical malfunction.

Auxiliary Building Addition

The SAFS pumps are located in the Auxiliary Building Addition adjacent to the south side of the Auxiliary Building. That portion of the Auxiliary Building Addition which encloses the pumps, the pumphouse, is a reinforced concrete structure designed to meet the seismic Class I criteria of Reference 6 and to protect the essential portions of the SAFS from the effects of tornados, including tornado missiles, and adverse environmental conditions. Air cooling and heating units in the pumphouse are designed to keep the room temperature suitable for operation of the SAFS. Access to the pumphouse is via a steel frame temporary storage building.

The Auxiliary Building Addition and temporary storage structure were reviewed to assure that their design or construction would not have an adverse impact on previously approved safety related structures or systems. Based on our review, we have determined that the installation of these structures imposes no adverse impact on existing safety related structures or systems. In fact, the pumphouse was built to more recent seismic design criteria than those identified in Reference 6.

Technical Specifications

The licensee has proposed operability and surveillance technical specification requirements for the SAFS (Reference 10). The staff reviewed the proposed specifications and concluded that some modifications were need to:

1. Better define the plant conditions when the SAFS (and the AFS) are required to be operable.

2. Provide acceptable time periods to repair an inoperable system flowpath.

3. Assure that the redundancy of the SAFS (and AFS) is maintained when feedwater is being obtained from the Service Water System.

Based on our review, we conclude that the proposed technical specifications, as modified, are acceptable

CONCLUSION

Based on our review of the information provided by the licensee, we conclude that (1) the SAFS is in conformance with the Commission's regulations as set forth in General Design Criteria (GDC) 19, "Control Room", GDC 44, "Cooling Water", GDC 45, "Inspection of Cooling Water Systems", GDC 46, "Testing of Cooling Water Systems", GDC 54, "Piping Systems Penetrating Containment", GDC 57, "Closed System Isolation Valves", and (2) the SAFS meets the guidelines of Regulatory Guides 1.26, "Quality Group Classification", and 1.62, "Manual Initiation of Protective Actions".

In addition, although several aspects of the SAFS and the Auxiliary Building Addition are being re-evaluated in the Systematic Evaluation Program, we have concluded that (1) the SAFS would acceptably mitigate the accident for which it was designed, (2) the installation of the SAFS and the Auxiliary Building Addition does not reduce existing safety margins for other safety related structures, systems and components, and (3) the proposed technical specifications for the SAFS, as modified by the staff, are acceptable. Therefore, the SAFS should be placed in an operable status to provide the additional plant protection for which it was designed.

Date: August 24, 1979

1. Report entitled, "Effects of Postulated Pipe Breaks Outside the Containment Buidling, Robert E. Ginna Nuclear Power Plant Unit 1", dated October 29, 1973.

2. RG&E letter, K. Amish to J. O'Leary, dated May 24, 1974.

3. RG&E letter, L. White to A. Schwencer, dated May 20, 1977, forwarding reports "Design Criteria Standby Auxiliary Feed System" , "Preliminary System Design Description for the Standby Auxiliary Feedwater System", and "Design Criteria for the Addition to the Auxiliary Building".

4. RG&E letter, L. White to D. Ziemann, dated July 28, 1978.

5. Systematic Evalulation Program, Status Summary Report, NUREG-0458, September 22, 1978.

6. Robert E. Ginna Nuclear Power Plant Unit No. 1, Final Facility Description and Safety Analysis Report.

7. Branch Technical Position ASB 3-1 appended to Standard Review Plan 3.6.1.

8. Branch Technical Position ASB 10-1 appended to Standard Review Plan 10.4.9.

9. RG&E letter, L. White to D. Ziemann, dated June 15, 1978.

10. Letter, LeBoeuf, Lamb, Leiby & MacRae to B. Rusche, dated February 1, 1977.

APPENDIX 2

DETAILED EVALUATION OF THE PRESSURE SHIELDING STEEL DIAPHRAGMS IN TURBINE BUILDING R. E. GINNA

SCOPE

The scope of this evaluation involves (a) assessment of the adequacy of the postulated design basis, (b) review of the ability of the "Pressure Shielding Steel Diaphragm"

DESCRIPTION OF STRUCTURE

The proposed pressure shielding steel diaphragm walls are being installed between the control building and the turbine building (i.e., adjacent to the control room, relay room and battery room) and between the diesel generator annex and the turbine building at the R. E. Ginna Nuclear Power Plant (Ginna). The new structures consist of horizontal steel beams (connected between existing steel columns) and vertical corrugated steel panels. The new steel beams provide support for the steel panel diaphragms. A detailed description of the modification can be found in the design criteria and engineering drawings provided by the Rochester Gas and Electric Corporation (the licensee) (References 1 and 5).

SUMMARY OF DESIGN LOADS:

Seismic Load (References 1 and 2):

A. Peak Ground Acceleration (a review of the definition of seismic input at the Ginna site currently is being conducted by the staff):

0.1g for Operating Basis Earthquake (OBE)

0.2g for Safe Shutdown Earthquake (SSE)

B. Regulatory Guide 1.60 Design Response Spectra were used.

C. Peak spectral acceleration used for design:

0.28g for OBE

0.55g for SSE

D. Horizontal and vertical seismic loads were applied simultaneously.

E. Damping ratio:

0.04 for OBE

0.07 for SSE

F. Equivalent static approach including "1.5" safety factor was used.

Pressure and temperature loads due to pipe break (reference 4):

Pa = 0.7 psi - Control Room

Pa = 1.14 psi - Diesel generator room, relay room and battery room

The temperature load "Ta" was converted to equivalent pressure load and combined with "Pa".

The combinations of the applied loads used for design are based on Standard Review Plan 3.8.4 (References 1 and 3).

Jet impingement effects were excluded since the high energy lines are located at a large distance from the diaphragm walls.

EVALUATION

A. The criteria used in the analysis and design of the new steel diaphragm walls to withstand the postulated loading conditions are in accordance with NRC Regulatory Requirements (Standard Review Plan 3.8.4, Regulatory Guides 1.29, 1.60, 1.61) and AISC, "Specification for the Design, Fabrication and Erection of Structural Steel for Buildings".

B. The postulated loading conditions (including dead loads, live loads, horizontal pressure loads, temperature load and seismic loads) and load combinations that may be imposed on the new walls during the service life-time of the plant conform to NRC - Standard Review Plan Section 3.8.4.

C. The equivalent static approach applied for the seismic analysis of structures is in accordance with NRC - Standard Review Plan Section 3.7.2, III-1b.

D. On the basis of the information provided by the licensee, we have concluded that the structural design provides reasonable assurance that the new walls will withstand the specified design conditions without impairment of structural integrity or performance of required safety functions and is therefore acceptable. This safety evaluation is based on the premise that the applied pressure and temperature loads on the steel diaphragm wall are caused by the postulated full diameter breaks in the 20" feedwater piping and in the 12" main steam pipe or postulated crack breaks in the 30" main steam line. The licensee provided the basis for these loads in their reports "High Energy Line Break Inside the Turbine Building," dated May 17 and July 6, 1979.

Date: August 24, 1979

References:

1. Design Criteria for Pressure Shielding Steel Diaphragm in Turbine Building, Revision 2, July 18, 1977.

2. Letter to D. L. Ziemann, NRC, from L. D. White, Jr., RG&E Corporation, August 25, 1978.

3. Letter to D. L. Ziemann, NRC from L. D. White, Jr., RG&E Corporation, October 11, 1978.

4. Summary of Meeting held on January 30, 1979.

5. Engineering Drawings - Turbine Building Pressurization Walls and Control Room Walls dated October 14, 1977, Drawing No. 04 4594 D-581-020, -021, -022, -023, -025, -026.

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-244

ROCHESTER GAS AND ELECTRIC CORPORATION

NOTICE OF ISSUANCE OF AMENDMENT TO PROVISIONAL OPERATING LICENSE

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 29 to Provisional Operating License No. DPR-18, issued to Rochester Gas and Electric Corporation (the licensee), which revised the Technical Specifications for operation of the R. E. Ginna Nuclear Power Plant (the facility) located in Wayne County, New York. The amendment is effective as of its date of issuance.

The amendment modifies the provisions of the Technical Specifications to incorporate the new Standby Auxiliary Feedwater System pumps, which relates to the result of the Commission's staff review of the licensee's analysis for high energy line breaks outside of containment.

The application for the amendment complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendment. Prior public notice of this amendment was not required since the amendment does not involve a significant hazards consideration.

The Commission has determined that the issuance of this amendment will not result in any significant environmental impact and that pursuant to 10 CFR §51.5(d)(4) an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with issuance of this amendment.

For further details with respect to this action, see (1) the Commission's letter to the licensee dated December 18, 1972, (2) the application for amendment dated February 1, 1977, and the licensee's letters dated February 6, 1978 and August 25, 1978, (3) Amendment No. 29 to License No. DPR-18, and (4) the Commission's related Safety Evaluation. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N. W., Washington, D. C. and at the Rochester Public Library, 115 South Avenue, Rochester, New York 14627. A copy of items (3) and (4) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Director, Division of Operating Reactors.

Dated at Bethesda, Maryland, this 24th day of August, 1979.

FOR THE NUCLEAR REGULATORY COMMISSION

Dennis L. Ziemann

Dennis L. Ziemann, Chief
Operating Reactors Branch #2
Division of Operating Reactors