NUREG-0944
Supplement No. 1

Safety Evaluation Report

related to the full-term operating license for R. E. Ginna Nuclear Power Plant

Docket No. 50-244

Rochester Gas and Electric Corporation

U.S. Nuclear Regulatory Commission

Office of Nuclear Reactor Regulation

October 1984

ABSTRACT

This report supplements the Safety Evaluation Report (NUREG-0944, October 1983) for the full-term operating license application filed by Rochester Gas and Electric Corporation for the R. E. Ginna Nuclear Power Plant (Docket No. 50-244). It has been prepared by the Office of Nuclear Reactor Regulation of the U.S. Nuclear Regulatory Commission. The facility is located in Wayne County, New York. Pending the favorable resolution of the items discussed in this report, the NRC staff concludes that the licensee can continue to operate the facility without endangering the health and safety of the public.

TABLE OF CONTENTS

ABBREVIATIONS

ACRS	Advisory Committee on Reactor Safeguards
ASLB	Atomic Safety and Licensing Board
ATWS	anticipated transient without scram
BTP	branch technical position
CCW	component cooling water
ECCS	emergency core cooling system
FTOL	full-term operating license
IE	Office of Inspection and Enforcement, NRC
IPSAR	integrated plant safety assessment report
JCO	justification for continued operation
LOCA	loss-of-coolant accident
NRC	Nuclear Regulatory Commission, U.S.
NSSS	nuclear steam supply system
OPS	overpressure protection system
POL	provisional operating license
PORV	power-operated relief valve
RCP	reactor coolant pump
RCS	reactor coolant system
RG&E	Rochester Gas and Electric Corporation
RHR	residual heat removal
SEP	Systematic Evaluation Program
SER	safety evaluation report
SSER	supplement to safety evaluation report
SWS	service water system
UPI	upper plenum injection
WOFA	Westinghouse optimized fuel assembly
WOG	Westinghouse Owners Group

1 INTRODUCTION AND DISCUSSION

1.1 Introduction

In October 1983, the Nuclear Regulatory Commission staff (NRC or staff) issued a Safety Evaluation Report (SER) on the R. E. Ginna Nuclear Power Plant, NUREG-0944, related to the full-term operating license application of Rochester Gas and Electric Corporation (RG&E or licensee) for conversion from a provisional operating license (POL) to a full-term operating license (FTOL). This document is the first supplement to that SER (SSER 1). It includes the report of the Advisory Committee on Reactor Safeguards (ACRS) (see Appendix D). This supplement also provides updated information available through September 1, 1984, on those subjects that have changed since the SER was issued and addresses issues that were the subject of Board Notifications applicable to Ginna.

There are a number of ongoing licensing actions for Ginna that are currently under staff review as noted in the SER and this supplement. The staff has determined that these items do not require resolution before the issuance of an FTOL and should not delay the POL-to-FTOL conversion process. All of these items will be addressed as routine operating reactor licensing actions after the FTOL is issued.

Each of the sections and appendices of this supplement is designated the same as the related portion of the SER. Appendix A is a continuation of the references of this safety review. Appendices B and C have been updated as necessary. Appendix D reproduces the ACRS report, which is addressed in Section 18.

At the time the SER was issued, there was an intervenor to the license conversion proceeding and it was expected that a public hearing would be held. By Memorandum and Order dated May 25, 1984, the Atomic Safety and Licensing Board (ASLB) directed each of the parties to file a report by August 15, 1984, indicating the status of the discovery process and a suggested schedule for prehearing and hearing activities. In a notice to the ASLB on July 24, 1984, the intervenor withdrew all of his outstanding contentions. As a result, the ASLB determined that there was no further need for evidentiary hearings in the proceeding and ordered the proceeding terminated.

The NRC Project Manager assigned to the FTOL review for Ginna is Mr. George Dick. Mr. Dick may be contacted by calling (301) 492-7215 or writing:

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3 DESIGN CRITERIA - STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS

3.2 Wind and Tornado Loadings

The licensee proposed to upgrade the facility (structural upgrade program) to withstand a 132-mph tornado in such a manner as to (1) achieve a safe hot shutdown with one completely protected train of equipment and have the capability to proceed to cold shutdown; (2) protect the reactor coolant pressure boundary, the main steam and feedwater lines, and the spent fuel assemblies; and (3) prevent accidents that could result in releases greater than the guidelines stated in 10 CFR Part 100.

The acceptance criteria established for the licensee's upgrade program correspond to a tornado event having a probability of exceedence of 10^{-5} per reactor year and encompasses issues raised for tornado wind loads, tornado missiles, the effect of block wall failure on main steam and feedwater piping, and portions of the seismic review. This is discussed in more detail in the conversion SER (NUREG-0944), the Integrated Plant Safety Assessment Report (IPSAR) (NUREG-0821), the licensee submittal of April 22, 1983, and the staff SER of August 22, 1983.

The development of the tornado hazard curve, structural margins for wind speeds beyond 132 mph, and the overall approach were discussed with the ACRS on April 5-7, 1984. At that meeting, the staff and the licensee expressed confidence in the accuracy of the hazard curve. Also, the licensee presented results from further analyses it has conducted. The analyses show that no instability or collapse that might affect components or systems needed for safe shutdown will occur for windspeeds up to approximately 200 mph. The ACRS has concluded that this is an adequate approach (Appendix D).

In its letter the ACRS recommended "that the NRC Staff consider further the measures proposed or needed to assure operability of the diesel generator during the reduced pressure transient accompanying a tornado." The licensee in a letter dated July 13, 1984, stated that an evaluation would be conducted to ensure that no operability restrictions exist. Any modifications which may be indicated will be incorporated with the structural upgrade program. The staff will review the evaluation as a separate licensing action. The results of the review will be discussed in a future staff safety evaluation specific to the operability of the diesel generator during a reduced pressure transient.

For reasons given above and those given in the IPSAR Supplement 1 (NUREG-0821, Supplement 1), dated August 1983, the staff concludes that the structural upgrade program is acceptable and additional protection would not be cost effective.

3.5 Effects of Pipe Break on Structures, Systems, and Components

In the SER, it was reported that the staff had evaluated the effects of pipe breaks to ensure that they would not cause the loss of needed functions of safety-related structures, systems, and components. Further, the staff performed the evaluation to ensure that the plant could be safely shut down in the

event of pipe breaks. Pipe breaks inside containment were evaluated under Systematic Evaluation Program (SEP) Topic III-5.A, and pipe breaks outside containment were evaluated under SEP Topic III-5.B.

In the SER, it was stated that the licensee proposed to eliminate high-energy steam heating lines in the relay room and air handling room to maintain a mild environment for equipment qualification purposes. The staff found the proposal acceptable. By letter dated August 30, 1984, the licensee informed the staff that the high-energy steam heating lines were removed during the 1984 refueling outage.

3.6 Seismic Design Considerations

(2) Piping System

The SER reported that as a result of SEP seismic review, NRC Office of Inspection and Enforcement (IE) Bulletins 79-02 and 79-14, and other NRC requirements, the licensee implemented a seismic upgrade program of all safety-related piping 2 inches in diameter and larger using current NRC licensing criteria. Upgrading involves both modifying and adding pipe supports. By letter dated August 30, 1984, the licensee informed the staff that the majority of the work has been accomplished. Some additional analysis and installation must still be performed. The licensee expects to complete any additional modifications during the 1985 and 1986 refueling outages.

(3) Mechanical Equipment

A total of 12 mechanical equipment items were initially sampled. Of the 12 items, 9 were determined to be adequate and 3 were determined to be inadequate. The four essential service-water pumps were determined to be not qualified with regard to structural and functional integrity because they lacked support near the suction of the pumps that overstressed the pump casing support. By letter dated August 30, 1984, the licensee informed the staff that modifications have been completed for upgrading the pumps.

The staff's initial evaluation included three safety-related tanks: component cooling surge tank, boric acid storage tank, and refueling water storage tank. The evaluation showed that the support of the component cooling surge tank needed to be upgraded and the refueling water storage tank required upgrading both with regard to support and structural integrity. By letter dated August 30, 1984, the licensee informed the staff that the component cooling surge tank has been upgraded.

Because two of the three tanks initially evaluated required upgrading, the licensee performed a seismic review of other safety-related tanks. On the basis of the staff's review, all safety-related field-erected tanks were found to be capable of withstanding the postulated seismic loadings except for three tanks: volume control tank, refueling water storage tank, and sodium hydroxide tank. It was reported in the SER that the issue of the volume control tank was resolved. The status of corrective actions for the remaining two tanks is discussed below:

(a) Refueling Water Storage Tank

The licensee had committed to perform a detailed evaluation for demonstrating the design adequacy of this tank. The licensee submitted the evaluation by letter dated September 13, 1983. The staff is currently reviewing it.

(b) Sodium Hydroxide Tank

According to the analysis performed by the licensee (letter dated April 11, 1983), this tank would not meet the acceptance criteria for the seismic analysis. The licensee had committed to upgrade this tank by the 1984 refueling outage. By letter dated August 30, 1984, the licensee informed the staff that the upgrading was completed.

(4) Control Room Electrical Panels

To demonstrate the structural integrity of panels (load path from an internally mounted element to anchorage and support systems), the licensee committed to conduct a low-impedance test for the main control board and, using the dynamic characteristics measured from the test, perform a seismic analysis to demonstrate its design adequacy. By letter dated January 9, 1984, the licensee transmitted a report entitled "Seismic Structural Evaluation of the Main Control Board." As a result of the evaluation, the licensee decided to enhance the structural integrity of the main control board. By letter dated August 30, 1984, the licensee informed the staff that the modifications were completed during the 1984 refueling outage.

(5) Electrical Cable Trays

The licensee is addressing this issue through the SEP Owners Group. The SER stated that for the evaluation of cable trays at Ginna, the licensee committed to provide a program plan by the end of 1983 based on the criteria to be proposed by the Owners Group.

As part of the SEP Owners Group studies, two reports dated April 29, 1983, and August 31, 1983, were submitted to the staff concerning the Owners Group efforts for seismic testing and analysis of cable tray and support configurations. By letter dated November 23, 1983, the staff responded with a set of questions. Most of these questions were discussed in a meeting of November 29, 1983, between NRC and the SEP Owners Group. By letter dated January 10, 1984, the licensee proposed a plan to identify and resolve, as necessary, outstanding plant-specific issues. The licensee indicated that the final schedule will be established after it has progressed toward resolving the staff questions and the staff accepts the SEP Owners Group report. The staff issued a draft evaluation of the report by letter dated July 5, 1984. The draft evaluation identifies issues and plant-specific actions needed to complete the resolution of the cable tray seismic integrity. A meeting between the SEP Owners Group and the staff is to be scheduled to discuss the evaluation.

4 REACTOR

4.1 Fuel System Design

By letter dated December 20, 1983, the licensee requested an amendment to the Technical Specifications which would permit the use of the Westinghouse optimized fuel assembly (WOFA). Beginning with Fuel Cycle 14 (June 14, 1984), the licensee began phasing-in the use of the WOFA with a reload containing approximately 17% WOFA. Eventually the reactor will be fueled with an all-WOFA core. The staff approved the use of WOFA in its Safety Evaluation for License Amendment No. 61, forwarded by a letter dated May 1, 1984.

5 REACTOR COOLANT SYSTEM AND CONNECTED SYSTEMS

5.4 Component and Subsystem Design

5.4.1 Reactor Coolant Pumps

5.4.1.2 Reactor Coolant Pump Seals

The Westinghouse Owners Group initiated a test program in June 1983 to determine the survivability of the reactor coolant pump (RCP) secondary seals under conditions wherein seal cooling is lost. The secondary seals were thought to be the weak components in the design of the RCP seal. Their failure under a loss-of-coolant condition could lead to a small-break loss-of-coolant accident (LOCA) resulting from failure of the complete RCP seal.

In the SER, it was reported that Westinghouse committed to supply the staff with additional information that would include a schedule for further testing and analysis. These subjects were discussed with the staff during a telephone conference call and submitted in a letter dated May 16, 1984.

The Westinghouse Owners Group reported that the tests of O-rings showed that they survived conditions beyond their design basis but exhibited unfavorable responses at high temperatures and high delta pressures. Analysis of the overall program demonstrated that reactor coolant system (RCS) leakage is constrained by the total seal package and its leak-off piping. Consequently, it was determined that the initial O-ring test program was overly conservative. The Westinghouse Owners Group is now undertaking further O-ring tests which will make use of the boundary conditions determined by the analysis. The tests are expected to be completed in the last half of 1984.

5.4.2 Steam Generators

5.4.2.5 Plugging Limits

The licensee has continued its efforts to develop and use steam generator tube sleeving as an alternative to plugging. By letter dated March 23, 1984, the licensee requested approval to install up to 10 tubesheet sleeves of a modified design. The modified design was based on efforts to improve inspectability and reduce the effect of postulated tube defects on the tubesheet region. The staff's approval and safety evaluation were transmitted by a letter dated May 9, 1984.

5.4.3 Residual Heat Removal System

Overpressure relief capacity is required for the residual heat removal (RHR) system when it is in operation; that is, when it is not isolated from the reactor coolant system. The overpressure protection system (OPS) fulfills this function and is required by Technical Specifications. There is no procedural requirement in the Technical Specifications that ensures that the OPS is in

service whenever the RHR system is in service. During cooldown, present procedures place the RHR system into service at 350°F and 360 psi, whereas the OPS is not required to be in service until the temperature is 330°F. The OPS must be in service when the reactor coolant system is less than 330°F, according to the Technical Specifications.

If the RHR system were placed into service before the OPS and a pressure transient occurred, there could be a pipe break in the RHR system outside containment.

In the SER it was reported that the licensee has proposed to revise the Technical Specifications to require that the OPS be in service whenever the RHR system is not isolated from the RCS. The basis for this requirement was to prevent the overpressurization of the RHR system that could lead to a LOCA outside containment. The staff found this acceptable.

As part of the amendment request (August 1, 1983) to the Technical Specifications, the licensee requested that the Technical Specifications be revised to require that the OPS be in service whenever the RHR system is not isolated from the RCS. This request will be acted upon in conjunction with the other items included in the licensee's amendment request.

10 CFR Part 50 (GDC 19 and 34), as implemented by SRP Section 5.4.7, Branch Technical Position RSB 5-1, and Regulatory Guide 1.139, requires that the plant can be taken from normal operating conditions to cold shutdown using only safety-grade systems (assuming a single failure) and either onsite or offsite power through the use of suitable procedures. The Ginna plant has safety-grade plant systems capable of safe shutdown under these conditions; however, the plant operating procedures rely on other nonsafety-grade systems and do not specify how the operator would accomplish cooldown if nonsafety-grade systems failed.

Nonsafety-grade systems may not be available following either a seismic event or a loss-of-power event. Insufficient information in operating procedures may increase the time required to shut down the reactor plant; for instance, if the nonsafety-grade air system failed, the steam atmospheric dump valves might be inoperable. The steam atmospheric dump valves would have to be opened manually if the backup nonsafety-grade nitrogen system also failed.

The licensee proposed to develop appropriate documented procedures for operating safety-grade systems and components to achieve cold shutdown if nonsafety-grade systems are unavailable. The staff found the proposal acceptable. The procedures are to be developed and implemented in coordination with Item 7 of Supplement 1 to NUREG-0737, "Upgrade Emergency Operating Procedures." By Confirmatory Order (letter dated June 12, 1984), the licensee committed to implement the upgraded emergency operating procedures by December 31, 1985.

6 ENGINEERED SAFETY FEATURES

6.2 Containment Systems

The containment isolation system of a nuclear power plant is an engineered safety feature that functions to allow the normal or emergency passage of fluids through the containment boundary while preserving the ability of the boundary to prevent or limit the escape of fission products to the environs that may result from postulated accidents. The safety review criteria for the containment isolation system have changed since Ginna began operation in 1971. SEP Topic VI-4 documents the deviations from current safety criteria as they relate to the containment isolation system. These differences are resolved in the IPSAR (NUREG-0821). The staff has required, or the licensee has proposed, a number of modifications to be performed on various containment isolation penetrations. The staff found the containment isolation system at Ginna acceptable, pending completion of these modifications. By letter dated August 30, 1984, the licensee informed the staff that procedural modifications to the containment isolation system were completed during the 1984 refueling outage. Hardware modifications will be completed before startup from the 1986 refueling outage.

6.3 Emergency Core Cooling System

Within this section, the SER addressed the issue of upper plenum injection (UPI). For two-loop units such as Ginna, the safety injection water is injected into the upper plenum rather than into the cold leg of the reactor coolant system (RCS). The staff was concerned that the emergency core cooling system (ECCS) models did not consider interaction between UPI water and steam.

By letter dated March 9, 1984, the staff advised the licensee of its plan for resolving the issue of UPI. The plan includes a safety evaluation documenting the status, results, and concerns following review of the Westinghouse and Exxon models for calculating the ECCS performance of Westinghouse two-loop plants. The licensee will be asked to provide a plan of action that responds to staff concerns.

8 ELECTRIC POWER SYSTEMS

8.4 Station Battery Capacity Test Requirements

As reported in the SER, the Ginna battery surveillance requirements are included in Section 4.6.2 of the plant's Technical Specifications. No periodic battery discharge test is required. Therefore, the Ginna plant does not comply with the current licensing requirements for station battery capacity tests. The staff's position in SEP Topic VIII-3.A was that the battery discharge test is a more severe test than the test that was being used and should be adopted.

In response to the staff's position, the licensee performed the battery discharge test during the spring 1982 refueling outage and has proposed appropriate changes to the Technical Specifications for battery testing by amendment request dated August 1, 1983. The request will be acted upon in conjunction with the other items included in the licensee's amendment request.

9 AUXILIARY SYSTEMS

9.1 Fuel Storage

By letter dated April 2, 1984, the licensee requested an amendment to the Technical Specifications to permit an expansion of the spent fuel pool storage capacity from 595 to 1016 spaces. The staff is reviewing this proposal and it is being handled as a separate licensing action. The results of the review will be discussed in a future staff safety evaluation specific to fuel storage.

9.2 Water Systems

The present Technical Specifications require that two of the four service-water-system (SWS) pumps and SWS loops be operational. However, if this plant were operating with the minimum number of required SWS pumps aligned to one bus and an accident occurred, the possibility exists that no SWS pump would be available. This is based on the assumption that one of the two emergency diesel generators would fail to start. Although the licensee's operation requires that an SWS pump be aligned to each bus, the staff's position is that the Technical Specifications should be made more explicit to ensure that the two operating SWS pumps are not serviced by the same diesel generator. The licensee has proposed a change to the Technical Specifications by amendment request dated August 1, 1983. The request will be acted upon in conjunction with the other items included in the licensee's amendment request.

During the SEP review, it was determined that the component cooling water (CCW) surge tank level is measured by a single transmitter with indication provided in the control room. This does not satisfy Section 4.20 of IEEE Std. 279-1971. The staff considered surge tank level an important parameter because it gives an anticipatory indication of possible loss of water and loss of system function which could affect the ability to safely shut down. The licensee proposed to add another transmitter to the surge tank with level alarms that would be independent of the present indicator in the control room.

By letter dated August 30, 1984, the licensee informed the staff that the modification has been completed.

9.6 Fire Protection

By letter dated December 27, 1983, the licensee requested an exemption from the requirements of 10 CFR 50.48(c)(4) that would extend implementation of the alternative shutdown system until startup from the 1986 refueling outage. The licensee informed the staff in the letter that alternative modifications had been developed which could be implemented with less disruption to the plant and phased so that portions of the plant will be in compliance with the rule earlier than would occur with installation of the staff-approved system. The licensee proposed interim post-fire safe-shutdown capability or interim fire-protection measures in support of its exemption request, for these noncompliance areas. The staff granted the exemption in a letter on May 10, 1984.

By letter dated January 16, 1984, the licensee submitted an alternative safe shutdown system. The staff is currently reviewing that submittal and it is being handled as a separate licensing action. The results of the review will be discussed in a future staff safety evaluation specific to fire protection.

17 QUALITY ASSURANCE

On January 5, 1984, Generic Letter 84-01 was issued to all holders of operating licenses, applicants for operating licenses, and holders of construction permits for power reactors regarding NRC use of the terms "important to safety" and "safety-related." The Generic Letter asked interested parties to contact the NRC about participating in a meeting to discuss the subject. The Utility Safety Classification Group, of which the licensee is a member, had provided its position earlier in a letter dated August 26, 1983.

By Attachement 2 to the letter dated November 4, 1983 (Required Actions Based on Generic Implications of Salem ATWS Events), the licensee provided the general system for classifying safety-related structures, systems, and components. The classifications are contained in an appendix to the Ginna Quality Assurance Manual. Further, an expanded listing of safety-related items was developed and included as part of the Ginna Quality Assurance Manual by revision dated December 30, 1983.

IE Bulletin 83-07 (July 22, 1983) informed nuclear power reactor facilities and fuel facilities about fraudulent products which may have been sold to nuclear industry companies by Ray Miller, Inc. during the years 1975-1979. By letter dated March 21, 1984, the licensee informed the NRC that, after reviewing the procurement activities for Ginna during the 1975-1979 time frame, it concluded that Rochester Gas and Electric Corp. (RG&E) had purchased no materials directly from Ray Miller, Inc. The licensee also stated that it has never installed or received equipment from other suppliers which contained materials provided by Ray Miller, Inc.

18 REPORT OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

During its 288th meeting, April 5-7, 1984, the ACRS completed its review of the application by RG&E for converting the provisional operating license for its R. E. Ginna Nuclear Power Plant to a full-term operating license (FTOL). This application was considered also during an ACRS subcommittee meeting in Washington, D.C., on November 16, 1983, and during the 283d ACRS meeting, November 17-19, 1983. Issues related to flood, severe wind, and earthquake hazards were reviewed in depth during meetings of the Subcommittee on Extreme External Phenomena on October 21-22, 1982, and April 4, 1984. Transcripts of the full committee and subcommittee meetings are available from Taylow Associates, Suite 1004, 1625 I Street, N.W., Washington, D.C. 20006. Copies of the transcripts are also available for review at the NRC Public Document Room, 1717 H Street, N.W., Washington, D.C., and at the Rochester Public Library, 115 South Avenue, Rochester, N.Y.

A copy of the ACRS report on the FTOL for the R. E. Ginna Nuclear Power Plant appears as Appendix D to this SER supplement. The report states that the ACRS believes there is reasonable assurance that the R. E. Ginna Nuclear Power Plant can continue to be operated at power levels up to 1,520 MWt under an FTOL without posing undue risk to the health and safety of the public.

APPENDIX A

CONTINUATION OF REFERENCES

Letter, December 17, 1982, D. G. Eisenhut (NRC) to All Licensees of Operating Reactors, Applicants for Operating Licenses, and Holders of Construction Permits, Subject: Supplement 1 to NUREG-0737, Requirements for Emergency Response Capability (Generic Letter No. 82-33).

---, April 11, 1983, J. E. Maier (RG&E) to D. M. Crutchfield (NRC), Subject: SEP Topics III-1, III-6, and IX-3.

---, April 22, 1983, J. E. Maier (RG&E) to D. M. Crutchfield (NRC), Subject: Structural Reanalysis Program, SEP Topics II-2.A, III-2, III-4.A, and III-7.B, R. E. Ginna Nuclear Power Plant.

---, April 29, 1983, R. M. Kacich (Northeast Utilities) to W. T. Russell (NRC) Subject: Shake Table Testing for Seismic Evaluation of Electrical Raceway Systems.

---, August 1, 1983, J. E. Maier (RG&E) to H. R. Denton (NRC), Subject: Licensee Amendment Request (various).

---, August 22, 1983, D. M. Crutchfield (NRC) to J. E. Maier (RG&E), Subject: Integrated Plant Safety Assessment Report (IPSAR), R. E. Ginna Nuclear Power Plant: Section 4.8, Wind and Tornado Loadings; Section 4.11, Tornado Missiles; and Section 4.17.1, Design Loads, Design Criteria and Load Combinations.

---, August 26, 1983, A. Early, Jr. (Counsel for Utility Safety Classification Group) to W. Dircks (NRC), Subject: Regulatory Terms "Safety-Related" and "Important to Safety."

---, August 31, 1983, R. M. Kacich (Northeast Utilities) to W. T. Russell (NRC), Subject: Analytical Techniques, Models and Seismic Evaluation of Electrical Raceway Systems.

---, September 13, 1983, J. E. Maier (RG&E) to D. M. Crutchfield (NRC), Subject: SEP Topic III-6, Seismic Qualification of Tanks, R. E. Ginna Nuclear Power Plant.

---, November 4, 1983, J. E. Maier (RG&E) to D. M. Crutchfield (NRC), Subject: Generic Letter 83-28, R. E. Ginna Nuclear Power Plant.

---, November 23, 1983, C. I. Grimes (NRC) to R. E. Schaffstall (KMC, Inc.), Subject: Seismic Qualification of Electrical Cable Trays.

---, December 20, 1983, J. E. Maier (RG&E) to D. M. Crutchfield (NRC), Subject: License Amendment Request: Use of Westinghouse Optimized Fuel Assembly as Reload Fuel, R. E. Ginna Nuclear Power Plant.

---, December 27, 1983, J. E. Maier (RG&E) to D. M. Crutchfield (NRC), Subject: 10 CFR 50.48 Exemption Application, R. E. Ginna Nuclear Power Plant.

---, January 4, 1984, J. E. Maier (RG&E) to D. M. Crutchfield (NRC), Subject: Seismic Evaluation of the Main Control Board.

---, January 5, 1984, D. G. Eisenhut (NRC) to All Holders of Operating Licenses, Applicants for Operating Licenses, and Holders of Construction Permits for Power Reactors, Subject: NRC Use of the Terms, "Important to Safety" and "Safety Related" (Generic Letter 84-01).

---, January 9, 1984, J. E. Maier (RG&E) to D. M. Crutchfield (NRC), Subject: SEP Topic III-6, Seismic Considerations, R. E. Ginna Nuclear Power Plant.

---, January 10, 1984, J. E. Maier (RG&E) to D. M. Crutchfield (NRC), Subject: Cable Tray Seismic Evaluation, R. E. Ginna Nuclear Power Plant.

---, January 16, 1984, J. E. Maier (RG&E) to D. M. Crutchfield (NRC), Subject: 10 CFR Part 50 Appendix R, Alternative Shutdown System, R. E. Ginna Nuclear Power Plant.

---, March 9, 1984, D. M. Crutchfield (NRC) to R. W. Kober (RG&E), Subject: Upper Head Injection.

---, March 21, 1984, R. W. Kober (RG&E) to T. E. Murley (NRC), Subject: IE Bulletin No. 83-07: Apparently Fraudulent Products Sold by Ray Miller, Incorporated, R. E. Ginna Nuclear Power Plant.

---, March 23, 1984, R. W. Kober (RG&E) to D. M. Crutchfield (NRC), Subject: Steam Generator Tubesheet Sleeves, R. E. Ginna Nuclear Power Plant.

---, April 2, 1984, R. W. Kober (RG&E) to H. R. Denton (NRC), Subject: License Amendment Request: Increase the Storage Capacity of the Spent Fuel Pool Storage Racks.

---, May 1, 1984, D. M. Crutchfield (NRC) to R. W. Kober (RG&E), Subject: Use of Westinghouse Optimized Fuel Assembly (OFA) as Reload Fuel.

---, May 9, 1984, D. M. Crutchfield (NRC) to R. W. Kober (RG&E), Subject: Steam Generator Sleeving - Tubesheet Sleeve.

---, May 10, 1984, D. M. Crutchfield (NRC) to R. W. Kober (RG&E), Subject: Schedular Exemption - Compliance with Appendix R to 10 CFR Part 50; Fire Protection for Nuclear Power Facilities.

---, May 16, 1984, J. J. Sheppard (WOG) to R. H. Vollmer (NRC), Subject: Explanation of the Westinghouse Owners Group Approach to the Investigation of RCP Seal Performance Following a Loss of All AC Power.

---, June 12, 1984, D. M. Crutchfield (NRC) to R. W. Kober (RG&E), Subject: Order Confirming Licensee Commitments on Emergency Response Capability (Generic Letter 82-33, Supplement 1 to NUREG-0737).

---, July 5, 1984, F. J. Miraglia (NRC) to R. E. Schaffstall (KMC, Inc.), Subject: Draft Evaluation of the SEP Owners Group Seismic Evaluation of Cable Trays.

---, July 13, 1984, R. W. Kober (RG&E) to D. M. Crutchfield (NRC), Subject: Structural Upgrade Program, SEP Topics II-2.A, III-2, III-4.A, and III-7.B, R. E. Ginna Nuclear Power Plant.

---, August 30, 1984, R. W. Kober (RG&E) to W. A. Paulson (NRC), Subject: FTOL Supplement, R. E. Ginna Nuclear Power Plant.

U.S. Nuclear Regulatory Commission, IE Bulletin 79-02, "Pipe Support Base Plate Designs Using Concrete Expansion Anchor Bolts," March 5, 1979; Rev. 1, June 1, 1979; Suppl. 1, August 17, 1979; Rev. 2, November 8, 1979.

---, IE Bulletin 79-14, "Seismic Analysis for As-Built Safety-Related Piping Systems," July 2, 1979; Rev. 1, July 17, 1979; Suppl., August 15, 1979; Suppl. 2, September 6, 1979.

---, IE Bulletin 83-07, "Apparently Fraudulent Products Sold by Ray Miller, Inc.," July 22, 1983.

---, NUREG-0821, "Integrated Plant Safety Assessment - Systematic Evaluation Program - R. E. Ginna Nuclear Power Plant," November 1982; Suppl. 1, August 1983.

APPENDIX B

THREE MILE ISLAND - LESSONS LEARNED REQUIREMENTS

The information included in this appendix provides an updated status of the various items. Items will be addressed in the body of this appendix only if there has been a change since the SER was issued. Updated versions of Tables B.1 (TMI items completed) and B.2 (TMI items completed, according to Rochester Gas and Electric Corporation (RG&E) but not yet reviewed by NRC) are also included.

There has been no change in Table B.3 (TMI items covered by Supplement 1 to NUREG-0737). The Confirmatory Order of June 12, 1984 commits the licensee to completion dates for those items.

I.A.1.3.2 MINIMUM SHIFT CREW

This position defines shift manning requirements for normal operation. The letter of July 31, 1980 from D. G. Eisenhut to all power reactor licensees and applicants set forth the interim criteria for shift staffing. Subsequent rule-making has codified the minimum shift crew in 10 CFR 50.54 and requires all licensees to meet the requirements by January 1, 1984.

Discussion

The licensee was in compliance with the rule on January 1, 1984. By letter dated April 3, 1984, the licensee requested an exemption until May 21, 1984. The circumstances necessitating the request involved NRC staff programmatic reviews of the licensee operator requalification program. As a result of the reviews, additional actions were required of the licensee involving further evaluation and testing of the licensed operators. In order to accomplish the evaluation and testing, it was necessary for more than the normal number of operators to be off shift. The licensee proposed compensating measures which were accepted by the staff. The exemption was granted on May 4, 1984. Following expiration of the exemption (May 21, 1984), the licensee has been in compliance with the rule.

II.B.1 REACTOR COOLANT SYSTEM VENTS

Each applicant and licensee shall install reactor coolant system (RCS) and reactor vessel head high point vents remotely operated from the control room. Although the purpose of the system is to vent noncondensable gases from the RCS which may inhibit core cooling during natural circulation, the vents must not lead to an unacceptable increase in the probability of a loss-of-coolant accident (LOCA) or a challenge to containment integrity. Since these vents form a part of the reactor coolant pressure boundary, the design of the vents shall conform to the requirements of Appendix A to 10 CFR 50, "General Design Criteria." The vent system shall be designed with sufficient redundancy that ensures a low probability of inadvertent or irreversible actuation.

Discussion

As described in RG&E's letter dated July 1, 1981, the reactor coolant head vent system has been installed on the Ginna reactor. The system is operable even though the fuses in the control/power circuits for the solenoid valves have been removed in the control room. Permission to operate the vents must be granted by the Plant Superintendent or the Technical Support Center Manager.

By letter dated September 28, 1983, the staff informed the licensee that the implementation, schedule, and requirement for a preimplementation review for RCS vents have been superseded by the requirements of 10 CFR 50.44(c)(3)(iii). All operating reactors, in order to provide the improved operational capability required by the rule, must have the RCS vents installed and operational, and procedures must be established and personnel trained in accordance with the schedule provided in the rule.

II.K.2.13 THERMAL MECHANICAL REPORT - EFFECT OF HIGH-PRESSURE INJECTION ON VESSEL INTEGRITY FOR SMALL-BREAK LOSS-OF-COOLANT ACCIDENT WITH NO AUXILIARY FEEDWATER

A detailed analysis shall be performed of the thermal-mechanical conditions in the reactor vessel during recovery from small breaks with an extended loss of all feedwater.

Discussion

Westinghouse (in support of the Westinghouse Owners Group) has performed analyses for generic Westinghouse plant groupings to address this issue. The generic study is applicable to the R. E. Ginna plant. The licensee sent a report of the study to the NRC by letter dated December 30, 1981.

By letter dated June 13, 1984, the staff informed the licensee that the information submitted adequately demonstrates reasonable assurance that vessel integrity is maintained for the event described by Item II.K.2.13, and that the requirements set forth in NUREG-0737, Item II.K.2.13, have been satisified.

II.K.3.1 INSTALLATION AND TESTING OF AUTOMATIC POWER-OPERATED RELIEF VALVE ISOLATION SYSTEM

All pressurized-water-reactor licensees should provide a system that uses the power-operated relief valve (PORV) block valve to protect against a small-break loss-of-coolant accident. This system will automatically cause the block valve to close when the reactor coolant system pressure decays after the PORV has opened. Justification should be provided to ensure that failure of this system would not decrease overall safety by aggravating plant transients and accidents.

Each licensee shall perform a confirmatory test of the automatic block valve closure system following installation.

Discussion

Implementation of this Action Plan item was modified in the May 1980 version of NUREG-0660. The change delays implementation of this item until after the

studies specified in TMI Action Plan for Item II.K.3.2 have been completed, if such studies confirm that the subject system is necessary. As discussed in Item II.K.3.2 (below), no further response is necessary.

II.K.3.2 REPORT ON OVERALL SAFETY EFFECT OF POWER-OPERATED RELIEF VALVE ISOLATION SYSTEM

(1) The licensee should submit a report for staff review documenting the various actions taken to decrease the probability of a small-break loss-of-coolant accident caused by a stuck-open power-operated relief valve (PORV) and show how those actions constitute sufficient improvements in reactor safety.

(2) Safety-valve failure rates based on past history of the operating plants designed by the specific nuclear steam supply system (NSSS) vendor should be included in the report submitted in response to (1) above.

Discussion

This Action Plan item called for documentation of modifications, if required, for automatic isolation of PORVs. An analysis performed for the Westinghouse Owners Group determined that automatic isolation of PORVs is not required. The results of the analysis and this conclusion were submitted by letter dated March 13, 1981.

The staff completed its review of the Westinghouse Owners Group submittal which the licensee endorsed for Ginna. The conclusions of the submittal were that "the concept of an automatic PORV block valve closure system, which closes the PORV isolation valves when lower pressure is sensed subsequent to a PORV failing to close, cannot be warranted on the basis of providing additional protection against a PORV LOCA." On that basis, the licensee proposed no modifications to provide automatic isolation of the PORVs in response to Item II.K.3.1.

After reviewing the submittal, the staff found that the requirements of NUREG-0737, Item II.K.3.2, are met with existing PORV, safety valve, and reactor high-pressure trip setpoints, and that an automatic PORV isolation system is not required for Ginna (September 27, 1983).

II.K.3.5 AUTOMATIC TRIP OF REACTOR COOLANT PUMPS DURING LOSS-OF-COOLANT ACCIDENT

Tripping of the reactor coolant pumps in case of a loss-of-coolant accident (LOCA) is not an ideal solution. Licensees should consider other solutions to the small-break LOCA problem (e.g., an increase in safety-injection flow rate). In the meantime, until a better solution is found, the reactor coolant pumps should be tripped automatically in case of a small-break LOCA. The signals designated to initiate the pump trip are discussed in NUREG-0623.

Discussion

Discussion of the background and scope of the issue as it pertains to Ginna are included in the SER and in Generic Letter 83-10d (February 8, 1983). The Westinghouse Owners Group (WOG) addressed this issue in letters OG-117

(March 9, 1984) and OG-110 (December 1, 1983). By letter dated April 10, 1984, the licensee endorsed the documents WOG submitted to the NRC. The staff is reviewing the documents. The review will be reported in a separate safety evaluation specific to the trip of reactor coolant pumps during a LOCA.

II.K.3.30 REVISED SMALL-BREAK LOSS-OF-COOLANT-ACCIDENT METHODS TO SHOW COMPLIANCE WITH 10 CFR PART 50, APPENDIX K

The analysis methods used by nuclear steam supply system vendors and/or fuel suppliers for small-break loss-of-coolant-accident (LOCA) analysis for compliance with Appendix K to 10 CFR Part 50 should be revised, documented, and submitted for NRC approval. The revisions should account for comparisons with experimental data, including data from the LOFT Test and Semiscale Test facilities.

Discussion

By letter dated January 19, 1982, RG&E presented the Westinghouse position that the small-break LOCA analysis model currently approved by the NRC for use on Ginna is conservative and is in conformance with Appendix K to 10 CFR Part 50. However, Westinghouse believes that improvement in the realism of small-break calculations is a worthwhile effort and has submitted its revised small-break LOCA analysis model (WCAP-10054) to address NRC concerns. The staff is currently reviewing Westinghouse's submittals and plans to issue a safety evaluation during the second quarter of Fiscal Year 1985.

REFERENCES

Letter, July 31, 1980, from D. G. Eisenhut (NRC) to All Power Reactor Licensees, Subject: Interim Criteria for Shift Staffing.

---, March 13, 1981, from R. W. Jurgensen (WOG) to J. R. Miller (NRC), Subject: WCAP-9804, "Probabilistic Analysis and Operational Data in Response to Item II.K.3.2 for Westinghouse NSSS Plants."

---, July 1, 1981, from J. E. Maier, (RG&E) to D. M. Crutchfield (NRC), Subject: NUREG-0737 Requirements.

---, December 30, 1981, from O. D. Kingsley (WOG) to H. R. Denton (NRC), Subject: WCAP-10019 "Summary Report on Reactor Vessel Integrity for Westinghouse Operating Plants."

---, January 19, 1982, from J. E. Maier (RG&E) to D. M. Crutchfield (NRC), Subject: Status of January 1, 1982, NUREG-0737 Items.

---, February 8, 1983, from D. G. Eisenhut (NRC) to All Licensees with Westinghouse (W) Designed Nuclear Steam Supply Systems (NSSSs) (Except Yankee Atomic Electric Company), Subject: Resolution of TMI Action Item III.K.3.5, "Automatic Trip of Reactor Coolant Pumps" (Generic Letter 83-10d).

---, September 27, 1983, from D. M. Crutchfield (NRC) to J. E. Maier (RG&E), Subject: NUREG-0737 Item II.K.3.1, Automatic PORV Isolation and II.K.3.2 - Report on PORVs.

---, September 28, 1983, from D. M. Crutchfield (NRC) to J. E. Maier (RG&E), Subject: NUREG-0737, Item II.B.1, Reactor Coolant System Vents.

---, December 1, 1983, OG-110, from J. Sheppard (WOG) to R. Mattson (NRC), Subject: Westinghouse Owners Group, Section I of Generic Letter 83-10c & d, WOG Report "Evaluation of Alternate RCP Trip Criteria."

---, March 9, 1984, OG-117, from J. Sheppard (WOG) to R. Mattson (NRC), Subject: Westinghouse Owners Group, Generic Letter 83-10c & d, WOG Report "Justification of Manual RCP Trip of SBLOCA Events."

---, April 3, 1984, from R. W. Kober (RG&E) to D. M. Crutchfield (NRC), Subject: Shift Staffing Requirements, R. E. Ginna Nuclear Power Plant.

---, April 10, 1984, from R. W. Kober (RG&E) to D. M. Crutchfield (NRC), Subject: Final Response to Generic Letter No. 83-10d, "Resolution of TMI Action Item II.K.3.5, Automatic Trip of Reactor Coolant Pumps," R. G. Ginna Nuclear Power Plant.

---, May 4, 1984, from D. M. Crutchfield (NRC) to R. W. Kober (RG&E), Subject: Exemption-Licensed Operator Staffing Rule 10 CFR 50.54(m)(2).

---, June 12, 1984, from D. M. Crutchfield (NRC) to R. W. Kober (RG&E), Subject: Order Confirming Licensee Commitments on Emergency Response Capability (Generic Letter 82-33 Supplement 1 to NUREG-0737).

---, June 13, 1984, from D. M. Crutchfield (NRC) to R. W. Kober (RG&E), Subject: NUREG-0737, Item II.K.2.13, "Thermal-Mechanical Report."

U.S. Nuclear Regulatory Commission, NUREG-0623, "Generic Assessment of Delayed Reactor Coolant Pump Trip During Small Break Loss-of-Coolant Accidents in Pressurized Water Reactors," November 1979.

---, NUREG-0660, "NRC Action Plan Developed as a Result of the TMI-2 Accident," May 1980; Rev. 1, July 1980.

---, NUREG-0737, "Clarification of TMI Action Plan Requirements," November 1980.

Westinghouse Electric Corporation, WCAP-10054, "Westinghouse Small Break ECCS Evaluation Model Using NOTRUMP Code," December 31, 1982.

Table B.1 TMI items completed

TMI Action Item	Shortened title	Date of NRC closeout letter
I.A.1.1	Shift Technical Advisor	1/12/82
I.A.1.2	Shift Supervisor Responsibilities	7/7/82
I.A.1.3.1	Shift Manning Overtime Limits	11/16/81
I.A.2.1	Immediate Upgrading of Reactor Operator and Senior Reactor Operator Training and Qualifications - Modify Training	7/2/82
I.A.2.3	Administration of Training Programs	10/31/80
I.C.2	Shift and Relief Turnover Procedures	7/7/80
I.C.3	Shift Supervisor Responsibility	7/7/80
I.C.4	Control Room Access	7/7/80
I.C.5	Feedback of Operating Experience	11/16/81
I.C.6	Verify Correct Performance of Operating Activities	11/16/81
II.B.2	Design Review of Plant Shielding and Environ mental Qualification of Equipment for Spaces/ Systems Which May Be Used in Postaccident Operations	5/23/84
II.B.3	Postaccident Sampling Capability	4/24/84
II.B.4	Training for Mitigating Core Damage	7/2/82
II.D.3	Valve Position Indication	5/11/81
II.E.1.1	Auxiliary Feedwater System Reliability	6/16/82
II.E.1.2	Auxiliary Feedwater System Initiation and Flow	8/18/82
II.E.3.1.1	Emergency Power for Pressurizer Heaters - Upgraded Power Supplies	7/7/80
II.E.3.1.2	Emergency Power for Pressurizer Heaters - Technical Specifications	5/11/81
II.E.4.1	Dedicated Hydrogen Penetrations	7/20/81

Table B.1 (Continued)

TMI Action Item	Shortened title	Date of NRC closeout letter
II.E.4.2.1-4	Improve Diverse Isolation	7/7/80
II.E.4.2.5	Containment Pressure Setpoint	1/13/82
II.E.4.2.6	Containment Purge Valves	12/15/82
II.E.4.2.7	Radiation Signal on Purge Valves	12/15/82
II.E.4.2.8	Containment Isolation Technical Specifications	5/11/81
II.F.1.1	Noble Gas Monitor	10/20/81
II.F.1.2	Iodine/Particulate Sampling	10/20/81
II.F.1.3	Containment High-Range Monitor	1/11/82
II.F.1.4	Accident Monitoring - Containment Pressure	5/10/84
II.F.1.5	Accident Monitoring - Containment Water Level	5/10/84
II.F.1.6	Accident Monitoring - Containment Hydrogen	5/10/84
II.G.1	Power Supplies for Pressurizer Relief Valves, Block Valves, and Level Indicators	5/11/81
II.K.1	IE Bulletins	5/28/80
II.K.2.19	Benchmark Analysis of Sequential Auxiliary Feedwater Flow	6/29/81
II.K.3.3	Reporting Safety Valve and Relief Valve Failures and Challenges	8/18/82
II.K.3.9	Proportional Integral Derivative Controller	8/25/81
II.K.3.10	Proposed Anticipatory Trip Modifications	8/25/81
II.K.3.12	Anticipatory Trip on Turbine Trip	8/25/81
II.K.3.17	Report on Outages of Emergency Core Cooling Systems Licensee Report and Proposed Technical Specifications	8/17/83
II.K.3.25	Power on Pump Seals	7/2/82
III.A.1.1	Emergency Preparedness, Short Term	10/31/80

Table B.1 (Continued)

TMI Action Item	Shortened title	Date of NRC closeout letter
III.A.2.1	Emergency Preparedness, Long Term	5/25/83
III.D.1.1	Primary Coolant Outside Containment	5/11/81
III.D.3.3	Inplant Radiation Monitoring	2/23/82
III.D.3.4	Control Room Habitability	4/11/83

Table B.2 TMI items completed according to Rochester Gas and Electric Corporation (RG&E) but not yet reviewed by NRC

TMI Action Item	Shortened title	Date of RG&E closeout letter
I.A.3.1	Review Scope and Criteria for Licensing Examinations	4/23/82
II.D.1.2	Relief Valve and Safety Test Programs	3/4/83
II.D.1.3	Block Valve Testing	6/11/82

APPENDIX C

UNRESOLVED SAFETY ISSUES

The information included in this appendix will update the status of the various issues. Issues will be addressed in this appendix only if their status has changed since the SER was issued. The reader is directed to the SER for discussion of the background of the issues and their applicability to Ginna.

There have been no changes to the tables.

C.4 Discussion of Tasks as They Relate to Ginna

A-2 Asymmetric Blowdown Loads on the Reactor Coolant System

By Generic Letter 84-04 (February 1, 1984), the licensee was informed that the staff has completed its review of Westinghouse topical reports and the letter report submitted to address asymmetric blowdown loads on the pressurized-water-reactor primary systems that result from a limited number of discrete break locations as stipulated in NUREG-0609, the staff's resolution of Unresolved Safety Issue A-2.

The staff concludes that an acceptable technical basis has been provided, so that the asymmetric blowdown loads resulting from double-ended pipe breaks in main coolant loop piping need not be considered as a design basis for the Westinghouse Owners Group plants, provided that two conditions are met.

Only the second of the two conditions applies to Ginna. It states that leakage-detection systems at the facility should be sufficient to provide adequate margin to detect the leakage from the postulated circumferential throughwall flow utilizing the guidance of Regulatory Guide 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems," with the exception that the seismic qualification of the airborne particulate radiation monitor is not necessary. At least one leakage-detection system with a sensitivity capable of detecting 1 gpm in 4 hours must be operable.

Pending receipt of a response from the licensee to the above condition, the staff concludes that there is reasonable assurance that Ginna can continue to operate without presenting undue risk to the health and safety of the public.

A-9 Anticipated Transients Without Scram

As a result of the Salem ATWS (anticipated transient without scram) events, the Commission published NUREG-1000, "Generic Implications of ATWS Events at the Salem Nuclear Power Plant." In a letter dated July 8, 1983 (Generic Letter 83-28), the staff identified the actions all licensees needed to take on the basis of NUREG-1000. These actions address issues related to reactor trip system reliability and general management capability. By letter dated November 4, 1983, the licensee responded to Generic Letter 83-28. The staff is currently reviewing the licensee response. The schedules for implementation

will be negotiated between the NRC Project Manager and RG&E consistent with the staff's goal of integrating new requirements and considering the unique status of each plant and the relative safety importance of the improvements, combined with all other plant programs. The staff will report on the resolution of these actions.

On the basis of the above considerations and subject to satisfactory resolution of the open items identified, the staff has concluded that there is reasonable assurance that Ginna can continue to operate without presenting undue risk to the health and safety of the public, before this generic issue has been resolved.

A-24 Environmental Qualification of Safety-Related Electrical Equipment

On April 17, 1984, the staff and the licensee met to resolve the outstanding environmental qualification issues for Ginna. The areas discussed were the deficiencies identified in the Franklin Research Center Technical Evaluation Report, the licensee's program for complying with the requirements set forth in 10 CFR 50.49, and the licensee's justification for continual operation (JCO) as required. There are no JCOs currently in effect. On the basis of the meeting, the licensee submitted a report on August 30, 1984, formally addressing the three areas identified above.

A-36 Control of Heavy Loads Near Spent Fuel

By letter dated December 22, 1980, the staff asked the licensee to determine the extent to which the guidelines of NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants," are met. The responses were to be made in two phases. RG&E responded to Phase I by letter dated February 1, 1982. Additional information was provided on March 2, 1983; October 12, 1983; and January 11, 1984. The staff reviewed the submittal and reported on it in a letter dated January 18, 1984. In that safety evaluation, the staff concluded that the licensee satisfied the guidelines in NUREG-0612, Sections 5.1.1 and 5.3, and that the review of Phase I was complete.

By letters dated March 26, 1984, and July 31, 1984, the licensee provided information regarding "Control of Heavy Loads (Phase II)." The staff will report on that submittal at a later date in a safety evaluation specific to control of heavy loads.

On the basis of the present controls placed on movement of heavy loads at the Ginna plant, including the area in the vicinity of spent fuel, and the additional effort to be made in clarifying load paths and procedures, the staff concluded that this issue is being adequately addressed for the Ginna plant and that operation can continue without presenting undue risk to the health and safety of the public.

C.5 References

Franklin Research Center (FRC), FRC TER C5257-178, "Equipment Environmental Qualification," C. Crane et al., March 1981.

Letter, December 22, 1980, D. G. Eisenhut (NRC) to All Operating Reactors, Subject: Control of Heavy Loads.

---, February 1, 1982, J. E. Maier (RG&E) to D. G. Eisenhut (NRC), Subject: Control of Heavy Loads.

---, March 2, 1983, J. E. Maier (RG&E) to D. M. Crutchfield (NRC), Subject: Supplemental Report Addressing Technical Evaluation Report, dated August 19, 1982, R. E. Ginna Nuclear Power Plant.

---, July 8, 1983, D. G. Eisenhut (NRC) to All Licensees of Operating Reactors, Applicants for Operating Licenses, and Holders of Construction Permits, Subject: Required Actions Based on Generic Implications of Salem ATWS Events (Generic Letter 83-28).

---, October 12, 1983, J. E. Maier (RG&E) to D. M. Crutchfield (NRC), Subject: Control of Heavy Loads, R. E. Ginna Nuclear Power Plant.

---, November 4, 1983, J. E. Maier (RG&E) to D. M. Crutchfield (NRC), Subject: Generic Letter 83-28, R. E. Ginna Nuclear Power Plant.

---, January 11, 1984, J. E. Maier (RG&E) to D. M. Crutchfield (NRC), Subject: Control of Heavy Loads, R. E. Ginna Nuclear Power Plant.

---, January 18, 1984, D. M. Crutchfield (NRC) to J. E. Maier (RG&E), Subject: Control of Heavy Loads (Phase I).

---, February 1, 1984, D. G. Eisenhut (NRC) to All PWR Licensees, Construction Permit Holders, and Applicants for Construction Permits, Subject: Safety Evaluation of Westinghouse Topical Reports Dealing With Elimination of Postulated Pipe Breaks in PWR Primary Main Loops (Generic Letter 84-04).

---, March 26, 1984, R. W. Kober (RG&E) to D. M. Crutchfield (NRC), Subject: Control of Heavy Loads Final Report, R. E. Ginna Nuclear Power Plant.

---, July 31, 1984, R. W. Kober (RG&E) to W. A. Paulson (NRC), Subject: Control of Heavy Loads, R. E. Ginna Nuclear Power Plant, Final Report.

---, August 30, 1984, R. W. Kober (RG&E) to W. A. Paulson (NRC), Subject: Environmental Qualification of Electrical Equipment, R. E. Ginna Nuclear Power Plant.

U.S. Nuclear Regulatory Commision, NUREG-0609, "Asymmetric Blowdown Loads on PWR Primary Systems, Resolution of Generic Task Action Plan A-2," January 1981.

---, NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants - Resolution of Generic Technical Activity A-36," July 1980.

---, NUREG-1000, "Generic Implications ATWS Events at the Salem Nuclear Power Plant," Vol. 1, April 1983; Vol. 2, August 1983.

APPENDIX D

REPORT OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS ON FULL-TERM OPERATING LICENSE FOR THE R. E. GINNA NUCLEAR POWER PLANT

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

April 9, 1984

Honorable Nunzio J. Palladino
Chairman
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Dr. Palladino:

SUBJECT: ACRS REPORT ON FULL-TERM OPERATING LICENSE FOR THE R. E. GINNA NUCLEAR POWER PLANT

During its 288th meeting, April 5-7, 1984, the Advisory Committee on Reactor Safeguards completed its review of the application by the Rochester Gas and Electric Corporation (Licensee) for conversion of the provisional operating license (POL) for its R. E. Ginna Nuclear Power Plant to a full-term operating license (FTOL). This application was considered also during a Subcommittee meeting in Washington, D.C. on November 16, 1983 and during the 283rd ACRS meeting, November 17-19, 1983. Issues related to flood, severe wind, and earthquake hazards were reviewed in depth during meetings of the Subcommittee on Extreme External Phenomena on October 21-22, 1982 and April 4, 1984. During our review, we had the benefit of discussions with representatives of the Licensee and the NRC Staff. We also had the benefit of the documents referenced. The Committee most recently discussed and reported on this plant in a letter dated August 18, 1982 relating to the Systematic Evaluation Program (SEP) review of the Ginna Plant.

The Ginna Plant received a POL in September 1969 and began commercial operation in December of the same year. The Licensee applied for an FTOL in a timely fashion in August 1972, but review of this application was deferred by the NRC Staff in 1975, along with several other FTOL reviews. In 1978, the Ginna Plant was included in Phase II of the SEP because much of the review needed for the FTOL was similar in scope to that for the SEP.

In the Committee's letter reporting on the results of the SEP as applied to the Ginna Plant, the ACRS indicated that its review of the FTOL would be deferred until the NRC Staff had completed its actions on the SEP issues that were still pending and on the Unresolved Safety Issue (USI) and TMI Action Plan items. The SEP issues have been resolved to the satisfaction of the NRC Staff in the manner reported in Supplement No. 1 to the Integrated Plant Safety Assessment Report for the Ginna Plant. The status of the USI and TMI Action Plan items for the Ginna Plant has been discussed by the NRC Staff in its Safety Evaluation Report related to the FTOL for the Ginna Plant.

Although all of the actions proposed or committed to as a result of the SEP review have not yet been completed, we believe that the procedures and schedules that have been agreed to are satisfactory. A large proportion of the TMI Action Plan items have been completed and those remaining are in a status acceptable to the NRC Staff, and to us. A similar situation exists with regard to those USI items for which a resolution has been reached by the NRC Staff.

The Licensee has proposed to modify the plant to decrease its vulnerability to tornado winds and missiles. These modifications will be based on a tornado having a design wind velocity of 132 mph. Modifications to the steel structures will be based on criteria that will ensure no significant yielding at wind speeds up to 132 mph, and no instability or collapse that might affect components or systems needed for safe shutdown at wind speeds up to about 200 mph. It appears from the Licensee's analyses that the cost of plant modifications would increase sharply if design basis tornadoes significantly higher than 132 mph were used. The NRC Staff believes that these planned modifications will upgrade the plant design such that tornadoes will not be a dominant contributor to the risk of core melt. We believe that this is an adequate approach, but recommend that the NRC Staff consider further the measures proposed or needed to assure operability of the diesel generator during the reduced pressure transient accompanying a tornado.

We concur with the process used by the NRC Staff and the Licensee to assure that the plant is adequately protected from the effects of external floods. The procedures used by the NRC Staff to evaluate the seismic adequacy of the plant are reasonable and are similar to procedures used in seismic reevaluation of other SEP plants.

We do not believe that any of the pending actions related to the SEP, USI, or TMI Action Plan items would be accelerated by withholding an FTOL at this time.

In connection with our review of the SEP, we have considered the operating experience at the Ginna Plant and have found nothing that would preclude granting an FTOL at this time. We have also reviewed the most recent Systematic Assessment of Licensee Performance (SALP) Report for the Ginna Plant, for the period June 1, 1982 through May 31, 1983, and note that all activities reviewed were classed in either Category 1 or 2. We find this encouraging.

The Committee believes that there is reasonable assurance that the R. E. Ginna Nuclear Power Plant can continue to be operated at power levels

up to 1520 MWt under a full-term operating license without undue risk to the health and safety of the public.

Sincerely,

Jesse C. Ebersole
Chairman

References:

1. Rochester Gas and Electric Corporation, "Final Safety Analysis Report, R. E. Ginna Nuclear Power Plant," Volumes 1-3 and Supplements 1-12
2. U. S. Nuclear Regulatory Commission, "Integrated Plant Safety Assessment, Systematic Evaluation Program, R. E. Ginna Nuclear Power Plant," USNRC Report NUREG-0821, dated December 1982 and Supplement 1 dated August 1983
3. Letter from H. Denton, Director, Office of Nuclear Reactor Regulation to P. Shewmon, Chairman, ACRS, dated September 17, 1982, Subject: Staff Response to the ACRS Report on the Systematic Evaluation Program Review of the R. E. Ginna Nuclear Power Plant
4. U.S. Nuclear Regulatory Commission, "Safety Evaluation Report Related to the Full-Term Operating License for R. E. Ginna Nuclear Power Plant," USNRC Report NUREG-0944, dated October 1983
5. U.S. Nuclear Regulatory Commission, "NRC Report on the January 25, 1982 Steam Generator Tube Rupture at R. E. Ginna Nuclear Power Plant," USNRC Report NUREG-0909, dated April 1982
6. Letter dated September 26, 1983 from T. Murley, NRC Regional Administrator, to John E. Maier, Rochester Gas & Electric Corp., Subject: Systematic Assessment of Licensee Performance (SALP) Report
7. Institute for Disaster Research, Texas Tech University, "A Methodology for Tornado Hazard Probability Assessment," prepared for USNRC by J. R. McDonald, NUREG/CR-3058, dated October 1983

NRC FORM 335 (2-84) NRCM 1102, 3201, 3202

U.S. NUCLEAR REGULATORY COMMISSION

BIBLIOGRAPHIC DATA SHEET

SEE INSTRUCTIONS ON THE REVERSE.

1. REPORT NUMBER (Assigned by TIDC, add Vol. No., if any)

NUREG-0944
Supplement No. 1

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Safety Evaluation Report related to the full-term operating license for R.E. Ginna Nuclear Power Plant

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Pertains to Docket No. 50-244

13. ABSTRACT (200 words or less)

The Safety Evaluation Report for the full-term operating license application filed by Rochester Gas and Electric Corporation for the R.E. Ginna Nuclear Power Plant was prepared by the Office of Nuclear Reactor Regulation of the U.S. Nuclear Regulatory Commission and issued in October 1983. This supplement includes the ACRS review and updates appropriate sections of the SER as required. The facility is located in Wayne County, Rochester, New York. Pending the favorable resolution of the items discussed in this report, the staff concludes that the facility can continue to be operated without endangering the health and safety of the public.

14. DOCUMENT ANALYSIS – a. KEYWORDS/DESCRIPTORS

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