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	RGN1	1	1			
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JOHN E. MAIER
Vice President

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October 19, 1982

Director of Nuclear Reactor Regulation
Attention: Mr. Dennis M. Crutchfield, Chief
Operating Reactors Branch No. 5
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Subject: Unresolved Safety Issue Status
R. E. Ginna Nuclear Power Plant
Docket No. 50-244

Dear Mr. Crutchfield:

This letter and the attached report are being provided in response to a letter of June 17, 1982 from Gus C. Lainas to John E. Maier, requesting the status of applicable Unresolved Safety Issues (USIs) for the R. E. Ginna Nuclear Power Plant. In that letter, the NRC requested that the following information be provided relative to each Unresolved Safety Issue:

(1) has the issue been resolved at Ginna
(2) if so, how has it been resolved; and
(3) if full resolution has not occurred (including implementation of necessary hardware, procedures, etc.), what interim measures have been taken to assure that continued operation would not pose an undue risk to the public.

The attachment to this letter addresses each USI identified in Mr. Lainas' letter. For each USI, a description of the problem based on NRC documentation is provided along with the RG&E status.

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We trust that our responses to these issues are sufficient to be used in the NRC staff's Safety Evaluation Report regarding the conversion of the Provisional Operating License for Ginna to a Full-Term Operating License.

Very truly yours,

John Maier

John E. Maier

Attachment

Sworn and subscribed to me on
this 19th day of October 1982

Lynn I. Hauck

LYNN I. HAUCK
NOTARY PUBLIC, State of N.Y., Monroe County
My Commission Expires March 30, 19.84

Unresolved Safety Issues

Status Report

R. E. Ginna Nuclear Power Plant
Docket No. 50-244

October 19, 1982

Unresolved Safety Issue A-1

Water Hammer

Description of Problem:

Water hammer occurs when the inertial properties of a piping system fluid are rapidly altered, causing the development of potentially damaging pressure pulses. Since 1971 there have been numerous incidents involving water hammers in BWRs and PWRs. Water hammers have involved steam generator feedrings and piping, the RHR system, ECC systems, and containment spray, service water, feedwater and steam lines. The incidents have been attributed to such causes as rapid condensation of steam pockets, steam driven slugs of water, pump startup with partially empty lines, and rapid valve motion. Most of the damage reported has been relatively minor, involving pipe hangers and restraints; however, there have been several incidents which have resulted in piping and valve damage. None of these incidents have involved the release of radioactivity to the environs.

RG&E Status:

The primary objective of this task was to resolve the potential for water hammer damaging the PWR steam generators and feedwater lines. A number of factors at Ginna Station reduce the likelihood of steam generator water hammers, such as limiting auxiliary feedwater flow to less than 150 gpm when steam generator levels are low and there is no safety requirement for more feedwater;

automatic start of auxiliary feedwater on loss of all feedwater, loss of offsite power, low low level in any one steam generator, and safety injection; and the fact that there is only a short length of feedwater piping between a steam generator and its loop seal. The likelihood of water hammer was further reduced by installation in 1979 of "J" tubes on the Ginna steam generator feedrings. The NRC's December 20, 1979 Safety Evaluation Report for Ginna relative to steam generator water hammer concluded that ". . . the means for reducing the potential for steam generator water hammer at this facility [Ginna] are adequate . . . and no further action is required of the licensee with regard to steam generator water hammer."

No problems associated with water hammer in other fluid system lines have been experienced, or are anticipated, at Ginna.

Based on the Ginna design, operating experience, and operating procedures, RG&E considers that this issue is properly addressed for the Ginna plant, and that operation can continue without undue risk to the health and safety of the public.

Unresolve Safety Issue A-2

Asymmetric Blowdown Loads on the Reactor Coolant System

Description of Problem:

In the event of a postulated LOCA at a reactor vessel nozzle, asymmetric loading on the reactor vessel, its supports, and internals could result from transient differential pressures in the reactor cavity. Such loading could potentially cause damage to the ECCS lines, control rods, other reactor coolant system components, and the fuel assemblies.

RG&E Status:

RG&E is an active participant in the Westinghouse A-2 Owners Group addressing this issue. The following Westinghouse Topical Reports, which are applicable to Ginna, have been submitted to the NRC, and are currently being evaluated by the NRC staff and their contractor EG&G:

WCAP 9558 through Rev. 2, 5/82
WCAP 9787 through Rev. 1, 5/81
WCAP 9749 6/80
WCAP 9570 6/80 and 10/79
WCAP 9748 6/80
WCAP 9662 2/80, 1/80
WCAP 9628 11/79

NUREG-0609, "Asymmetric Blowdown Loads or PWR Primary Systems, Resolution of Generic Task Action Plan A-2," January 1981, has been published by the NRC. This document defines implementation criteria developed as part of A-2.

Although not yet issued by the NRC, RG&E expects that a final acceptance of the Westinghouse A-2 Owners Group and RG&E analyses will complete all open issues related to A-2. The analyses have already been discussed with the Advisory Committee on Reactor Safeguards.

Based on the analyses performed as part of the Westinghouse A-2 Owners Group, RG&E considers that the issue of Asymmetric Blowdown Loads has been adequately addressed for the Ginna plant, and that operation can continue without undue risk to the health and safety of the public.

Unresolved Safety Issue A-3

Steam Generator Tube Integrity

Description of Problem:

This issue addresses the capability of steam generator tubes to maintain their integrity during normal operation and under accident conditions, with adequate safety margins. PWR steam generator tubing in some plants has experienced tube wall thinning (wastage), intergranular attack, stress-corrosion cracking, and denting.

RG&E Status:

Rochester Gas and Electric replaced the original phosphate secondary side water chemistry treatment with an all-volatile treatment in November 1974 and added full flow condensate polishing demineralizers in 1978. At present, less than 5% of the tubes in each steam generator have been plugged. In addition, 21 tubes in the B steam generator have been sleeved. The primary reasons for tube repair have been wastage and crevice intergranular attack. As a result of the change in chemistry, wastage no longer appears to be occurring.

RG&E's present program of steam generator tube inspections provides for eddy current tests of the tubes, tube sheet water lancing, and crevice cleaning if determined to be necessary. Further, RG&E has proposed a sleeving program to install sleeves

as a preventive measure on those steam generator tubes considered most susceptible to crevice intergranular attack.

On January 25, 1982, Ginna Station experienced a steam generator tube rupture. The description of the event, and the restart Safety Evaluation Report, are NUREG-0909 and NUREG-0916, respectively.

Based on the inservice inspection and testing being performed on the Ginna steam generators, the existence of the metal impact monitoring system installed on the steam generators, and the proposed preventive sleeving program, RG&E considers that the subject of Steam Generator Tube Integrity is being adequately addressed for the Ginna plant, and that operation can continue without undue risk to the health and safety of the public.

Unresolved Safety Issue A-9

Anticipated Transients Without Scram (ATWS)

Description of Problem:

During operation of a nuclear power plant, key parameters are monitored and used to actuate safety systems that initiate shutdown (SCRAM) of the reactor. For a number of years there has been concern that, following a temporarily abnormal operation condition, or "anticipated transient," a failure could occur in the systems required to insert the control rods into the reactor, and a resultant scram might not occur. Failure to scram during or following an anticipated transient would increase the severity of the transient, and could cause fuel damage.

RG&E Status:

The Technical Report on Anticipated Transients Without Scram for Water-Cooled Power Reactors, WASH-1270, discussed the probability of an ATWS event and an appropriate safety objective for these events. WCAP 8404, "ATWT Analysis for Westinghouse PWR's with 44 Series Steam Generators," was released in September of 1974. Following review of this report, as well as the many other vendor reports describing the analysis models and results, the NRC staff published, in late 1975, its status report on each vendor analysis including detailed guidelines on analysis models, and ATWS safety objectives.

Since the publication of the 1975 status reports, additional information relevant to ATWS has been developed by the industry and the Reactor Safety Study Group. Based on review of these reports and discussions with vendors, an NRC report on "Anticipated Transients Without Scram for Light-Water Reactors," NUREG-0460, Volumes 1 and 2, was published in April 1978. Since the issuance of Volumes 1 and 2, additional safety and cost information and new insights were developed on the general subject of quantitative risk assessment. Based on these considerations, the NRC staff issued a new report, Volume 3 to NUREG-0460, dated December 1978. Volume 3 considered various alternative plant modifications for ATWS ranging from none to those needed to satisfy the proposed licensing criteria for new plants in NUREG-0460, Volumes 1 and 2. The staff assessed the corresponding degrees of assurance of safety achieved from these alternative modifications. In Volume 3, the staff also suggested plant modifications on the basis of the plant design and age. In order to confirm the staff judgement on the adequacy of these designs, the staff issued requests for industry to supply the necessary generic analyses. Generic Westinghouse responses, applicable to Ginna, were presented to the NRC by reports dated June 8, 1979, and December 30, 1979, "Anticipated Transients Without Scram for Westinghouse Plants." In NUREG-0460, Volume 4, issued in March 1980 for public comment, the NRC staff reviewed the industry responses. It was concluded that the necessary verification of the adequacy of the proposed design changes had not been provided. The NRC staff thus proposed that early improvements in safety should be provided, and any

additional requirements should be considered under the staff recommended rulemaking. The NRC has reviewed the industry and the ACRS comments in Volume 4, and has published a proposed rule for resolution of ATWS.

For Ginna specifically, it is not anticipated that any major hardware modifications will be required. RG&E is working with the Westinghouse Owners Group to develop guidelines for new Emergency Operating Instructions (EOI's), which will be modified into plant-specific Emergency Procedures. These will include instructions for mitigating an ATWS event.

Based on the extensive analysis to date showing no adverse consequences, the extremely low probability of an ATWS event, and the development of emergency procedure guidelines, RG&E considers that the ATWS event is being adequately addressed for the Ginna plant, and that operation can continue without undue risk to the health and safety of the public.

Unresolved Safety Issue A-11

Materials Toughness

Description of Problem:

Because the possibility of failure of nuclear reactor pressure vessels designed to the ASME Boiler and Pressure Vessel Code is remote, the design of nuclear facilities does not provide protection for reactor vessel failure. However, as plants accumulate more and more service time, neutron irradiation reduces the material fracture toughness and initial safety margins.

RG&E Status:

10CFR50, Appendices G and H, require that compliance with minimum fracture toughness requirements be demonstrated, and that a materials surveillance program to monitor changes in the fracture toughness properties of ferritic materials in the reactor vessel beltline region be maintained. This issue was discussed during the Systematic Evaluation Program review of SEP Topic V-6, Reactor Vessel Integrity, in NUREG-0569, "Evaluation of the Integrity of SEP Reactor Vessels." Based on the recommendations of that report, RG&E committed to provide evaluation of the next surveillance capsule, including a complete chemical analysis of the capsule to the NRC for review. The capsule was removed from the reactor in 1980, and was shipped to our contractor, Westinghouse, in 1981. Preliminary charpy results were transmitted to the NRC by letter dated October 6, 1981. Remaining analyses are being completed now and will be submitted soon.

Results to date indicate virtually no change in reactor vessel material properties from previous capsule results. The RT_{NDT} at 30 foot-pounds of energy is approximately 125°F, well below that considered of concern by Regulatory Guide 1.99.

Based on these acceptable results, and the May 3, 1982 letter from Dennis M. Crutchfield to John E. Maier which considered the SEP Topic V-6 relative to reactor vessel integrity to be complete, RG&E considers this Unresolved Safety Issue to be complete for Ginna.

Unresolved Safety Issue A-12

Fracture Toughness of Steam Generator and RCS Pump Supports

Description of Problem:

During the course of the licensing action for North Anna Power Station Units No. 1 and 2, a number of questions were raised as to the potential for lamellar tearing and low fracture toughness of the steam generator and reactor coolant pump support materials for those facilities. The toughness of one of the steels (A-572) used was relatively poor at an operating temperature of 80°F. Since similar materials and designs are used at other facilities, generic concerns were raised. It became necessary to reassess the fracture toughness of the steam generator and reactor coolant pump support materials for all PWRs.

The NRC reported a technical study (Appendix C to NUREG-0577, "Potential for Low Fracture Toughness and Lamellar Tearing a PWR Steam Generator and Reactor Coolant Pump Supports") conducted by Sandia Laboratories, which revealed that no documentation exists describing inservice failures resulting from lamellar tearing.

RG&E Status:

The materials of construction of the steam generators' and reactor coolant pumps' supports have been determined to be different from those used at the North Anna Station. RG&E's submittal relative to the support material was provided in a report trans-

mitted by letter dated June 26, 1978. It was concluded that adequate fracture toughness exists for the supports at Ginna Station. This report also reviewed the parameters which affect the potential for lamellar tearing in weldments -- steel quality, steel fabrication practice, weld joint design, grade of filler material, weld dimensions, and post-weld heat treatment. It was concluded that lamellar tearing would not be a problem for the Ginna design and installation.

Based on the review of the supports at Ginna, as described in the June 26, 1978 report, which showed adequate fracture toughness and resistance against lamellar tearing, plus the fact that RG&E is continuing to monitor developments in these areas, it is considered that this issue is being adequately addressed for the Ginna Station, and that operation can continue without undue risk to the health and safety of the public.

Unresolved Safety Issue A-17

Systems Interaction in Nuclear Power Plants

Description of Problem:

Due to the complex nature of the design of nuclear power plants, numerous engineering disciplines must be coordinated and systematically merged to produce an operating plant. Initial and subsequent designs undergo exhaustive specific and interdisciplinary review and evaluation to ensure safety is not adversely affected by the interaction of various systems. In conjunction with the design reviews conducted by the Architect Engineer and the Utility, the NRC also performs an independent review of system interactions. However, there remain questions regarding both the supporting role that systems play and the effect that one system can have on another. Examples of such system interactions include potential failures as a result of pipe breaks, fire, environmental effects, and seismically-induced motion. This USI was divided into two phases by the NRC.

Phase I was structured to identify areas where interactions are possible between and among systems that have the potential of negating or seriously degrading the performance of safety functions. Also, Phase I was to identify areas where NRC review procedures may not have properly accounted for these interactions.

The anticipated Phase II program will not be pursued as a USI. Phase II, which was originated to take specific corrective measures in areas where the Phase I shows a need, will be performed under TMI Action Plan Item II.C.3, Systems Interaction (reference: NUREG-0606, November 16, 1981).

RG&E Status:

The common-mode effects of various postulated external events, as well as in-plant events and failures, upon safety-related structures, systems, and components in order to ensure safe shutdown capability, have been extensively studied for the Ginna plant. These studies have been made both as a result of the Systematic Evalution Program (SEP) and the TMI Action Plan items. Areas most recently studied include the effects of seismic events, pipe breaks, internal and external flooding, wind and tornado loadings, internal missiles, and site hazards. Also, the RG&E fire protection study, together with our proposed course of action, provides substantial assurance that separation and independence of safety-related systems at Ginna are provided.

Based on the extensive nature of reviews to ensure safe shutdown capability under various common-mode events, RG&E considers that this issue is being adequately addressed, and that the Ginna plant can be safely operated without undue risk to the health and safety of the public.

Unresolved Safety Issue A-24

Environmental Qualification of Safety-Related Electrical Equipment

Description of Problem:

The evolutionary process of developing environmental qualification requirements and the case-by-case implementation of them has resulted in a diversity of equipment installed in nuclear plants, and different levels of documentation of the extent to which equipment is environmentally qualified. In an effort to further standardize the qualification methods and documentation, Generic Task A-24 was developed. Issuance of NUREG-0588 by the NRC in July 1981 completed this unresolved safety issue. For operating reactors such as the Ginna plant, the "DOR Guidelines," transmitted to RG&E by letter dated February 15, 1980, provide the basis for environmental qualification requirements.

RG&E Status:

By letter dated September 19, 1980, the NRC transmitted a Revised Order for Modification of License, effective immediately, directing that information regarding the environmental qualification of safety-related electrical equipment be submitted by November 1, 1980. Specifically, the NRC ordered that the submittal of information fully and completely respond to the NRC Staff's requests transmitted by letters dated March 6, 1980 and March 28, 1980.

Franklin Research Center, under contract to the NRC, reviewed the RG&E responses, and provided an assessment via Draft Interim Technical Evaluation Report FRC Project C5257-178, dated August 20, 1980. RG&E provided all requested information, as well as a response to the FRC Report, by letter and report dated October 31, 1980. Additional reviews by FRC resulted in a June 1, 1981 Safety Evaluation Report, with attached FRC TER C5257-178 (March 18, 1981). RG&E's response to this SER, dated September 4, 1981 provided for a replacement program for several safety-related components to increase the margins of qualification and to provide additional qualification documentation where necessary. A final commitment letter of June 10, 1982 revised this program to be consistent with the schedule stated in draft rule 10CFR 50.49, Section (h).

Based on RG&E's commitment to meet the regulatory criteria provided in the draft rule, and the acceptability of the presently installed equipment to withstand adverse environmental effects, RG&E considers that operation of the Ginna plant can continue without undue risk to the health and safety of the public.

Unresolved Safety Issue A-26

Reactor Vessel Pressure Transient Protection

Description of Problem:

Over the years there have been several reported incidents of pressure transients in PWRs which have exceeded the pressure/ temperature limits of the reactor vessels involved. Most events occurred while the plant was in a solid water condition, normally during startup or shutdown operations and at relatively low reactor vessel temperatures.

The causes of these overpressurizations were grouped into the following general categories: personnel error, procedural deficiencies, component random failures, and spurious valve actuation. The resultant pressure transient was the result of either a mass input (charging pumps, safety injection pumps and accumulators) or a thermal expansion of the primary fluid, typically from heat input from the steam generator.

In light of the frequency of these transients and the decreasing reactor vessel toughness with age (due to increased neutron fluence) the NRC adopted this task to develop methods to prevent and minimize the effects of reactor vessel overpressurization.

RG&E Status:

Rochester Gas and Electric installed a Reactor Vessel Low Temperature Overpressure Protection System during the 1978 refueling outage. The Technical Specification changes, and the Safety Evaluation Report accepting this system, were issued on April 18, 1979. This "Unresolved Safety Issue" is considered complete for the Ginna facility.

Unresolved Safety Issue A-31

Residual Heat Removal Requirements

Description of Problem:

The safe shutdown of a nuclear power plant following an accident not related to a Loss of Coolant Accident (LOCA) has been typically interpreted as achieving a "hot standby" condition (i.e., the reactor is shut down, but system temperature and pressure are still at or near normal operating values). Considerable emphasis has been placed on the hot standby condition of a power plant in the event of an accident or other abnormal occurrences. A similar emphasis has been placed on long-term cooling, which is achieved by the Residual Heat Removal (RHR) system. The RHR system starts to operate when the reactor coolant pressure and temperature are substantially lower than their hot standby condition values. However, there was only limited review of the transient conditions of getting from hot shutdown to cold shutdown conditions.

RG&E Status:

Safe shutdown, including maintenance of hot standby, cooldown, and cold shutdown operation, was the emphasis of review during the SEP evaluation of topics V-10.B, V-11.B, and VII-3. The review of this capability at Ginna was documented in the NRC's Safety Evaluation Report dated September 29, 1981. The only items requiring changes, as noted in NUREG-0821, Draft Integrated Plant Safety Assessment Report, dated May 1982 are:

a) an evaluation of the operating procedures to determine if additional guidance is required for control room personnel to effect cold shutdown using only safety-related equipment, and

b) a Technical Specification change to place the Low Temperature Overpressure Protection System (OPS) in effect prior to the use of the Residual Heat Removal System.

The NRC noted that sufficient capability to effect cold shutdown using only safety-related equipment did exist at the facility; only the procedures needed review. Also, by procedure, the OPS is put into effect prior to use of the RHR System. Further, the Sandia Laboratories Probabilities Risk Assessment Study for these two proposed changes (Appendix D of NUREG-0821) showed them to be of low safety significance.

Nevertheless, RG&E has committed to make both of these changes. Based on this commitment, and the relatively low safety priority of these two items, it is considered that this USI is being adequately addressed for the Ginna plant, and that operation can continue without undue risk to the health and safety of the public.

Unresolved Safety Issue A-36

Control of Heavy Loads Near Spent Fuel

Description of Problem:

Overhead handling systems (cranes) are used to lift heavy objects in the vicinity of spent fuel in PWRs and BWRs. If a heavy object, e.g., a spent fuel shipping cask or shielding block, were to fall or tip onto spent fuel in the storage pool or the reactor core and damage the fuel, there could be a release of radioactivity to the environment and a potential for radiation overexposure to inplant personnel. If many fuel assemblies are damaged, and the damaged fuel contained a large amount of undecayed fission products, radiation releases to the environment could exceed 10 CFR Part 100 guidelines.

Additionally, a heavy object could fall on safety-related equipment and prevent it from performing its intended function. If equipment from redundant shutdown paths were damaged, safe shutdown capability may be defeated.

RG&E Status:

The NRC requested, by letter dated December 22, 1980, that licensees make a determination of the extent to which the guidelines of NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants," are met. RG&E responded to this request by letter dated February 1, 1982. The auxiliary building crane movement over the

Spent Fuel Pool is limited by a system of electrical interlocks except for a small portion of the southeast corner, and a narrow strip on the north side. Administrative procedures limit travel in these aras of the pool. A review by Franklin Research Center, draft Technical Evaluation Report C5257-444, transmitted by NRC letter of August 19, 1982, is presently being evaluated by RG&E. In that report, FRC recommends that additional administrative clarifications in load handling procedures and more explicit marking of load paths be pursued.

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

Unresolved Safety Issue A-40

Seismic Design Criteria Short-Term Program

Description of Problem:

The seismic design process required by current NRC criteria includes the following sequence of events.

a. Define the magnitude or intensity of the earthquake which will produce the maximum vibratory ground motion at the site (the safe shutdown earthquake)

b. Determine the free-field ground motion at the site that would result if the SSE occurred.

c. Determine the motion of site structures by modifying the free-field motion to account for the interaction of the site structures with the underlying foundation soil.

d. Determine the motion of the plant equipment supported by the site structures.

e. Compare the seismic loads, in appropriate combination with other loads, on structures, systems, and components important to safety, with the allowable loads.

While this seismic design sequence includes many conservative factors, certain aspects of the sequence may not be conservative for all plant sites. At present it is believed that the overall sequence is adequately conservative. The objective of this program is to investigate selected areas of the seismic design sequence to determine their conservatism for all types of sites, to investigate alternate approaches to parts of the design sequence, to quantify the overall conservatism of the design sequence, and to modify the NRC criteria in the Standard Review Plan if changes are found to be justified. In this manner this program will provide additional assurance that the health and safety of the public is protected, and if possible, reduce costly design conservatisms by improving (1) current seismic design requirements, (2) NRC's capability to evaluate the adequacy of seismic design of operating reactors and plants under construction, and (3) NRC's capability to quantitatively assess the overall adequacy of seismic design for nuclear plants in general.

RG&E Status:

As a part of Systematic Evaluation Program (SEP) Topic III-6, Seismic Design Considerations, an extensive effort was made to evaluate and document the seismic design basis of Ginna Station. This included an evaluation by the NRC of the site specific response spectra, which was documented in a letter from the NRC to all SEP Owners dated June 8, 1981, which confirmed the conservatism of the original design basis. Additional seismic criteria were defined by NUREG/CR-0098, which showed that the

damping criteria of Regulatory Guide 1.61 were generally overly conservative, and in NUREG/CR-1717, "Soil-Structure Interaction Methods." Further, a Senior Seismic Review Team was formed to evaluate the seismic capability of safety-related structures, systems, and components. The results of that review for Ginna were documented in NUREG/CR-1821.

RG&E has made commitments to increase the seismic safety margins of certain structures, systems, and components, as documented in the draft Integrated Plant Safety Assessment Report, NUREG-0821. The significant safety margin available in the original plant specifications, together with the programmatic seismic upgrade of certain safety-related items at Ginna, confirms that this issue is adequately addressed for the Ginna plant, and that plant operation can continue without undue risk to the health and safety of the public.

Unresolved Safety Issue A-43

Containment Emergency Sump Reliability

Description of Problem:

Following a loss of coolant accident (LOCA) in a PWR, water flowing from the break in the primary system would collect on the floor of containment. During the injection mode, water for core cooling and containment spray is drawn from a large supply tank. When the water reaches a low level in the tank, pumps are realigned to draw from the containment. This is called the recirculation mode wherein water is drawn from the containment floor or sump and pumped to the primary system or containment spray headers. This program addresses the safety issue of adequate sump or suppression pool function in the recirculation mode. It is the objective of this program to develop improved criteria for design, testing and evaluation which will provide better assurance that emergency sumps will function to satisfy system requirements.

The principal concerns are somewhat interrelated but are best discussed separately. One deals with the various kinds of insulation used on piping and components inside containment. The concern is that break-initiated debris from the insulation could cause blockage of the sump or otherwise adversely affect the operation of the pumps, spray nozzles, and valves of the safety systems.

The second concern deals with the hydraulic performance of the sump as related to the operation of safety systems supplied from the sump. Preoperational tests have been performed on a number of plants to demonstrate operability in the recirculation mode. Adverse flow conditions have been encountered requiring design and procedural modifications to eliminate them. These conditions, air entrainment, cavitation, and vortex formation, are aggravated by blockage. If not avoided, the effects could result in loss of net positive suction head (NPSH), and pump damage, in the long term cooling phase following a LOCA.

RG&E Status:

RG&E has been following the sump hydraulic experiments being conducted at the Alden Research Laboratories for a wide range of sump designs and adverse plant effects. Data have shown that air ingestion is lower than had been expected, and that there has thus been virtually no adverse effect on pump performance under all except the most conservatively postulated conditions (NUREG/CR-2792). A plant-specific analysis of the Ginna emergency procedure for switching from the Refueling Water Storage Tank to the sump, performed as part of SEP Topic VI-7.B, ESF Switchover, disclosed that RHR flows from the sump would be quite low, and that significant NPSH margins are available.

RG&E also participated in an NRC survey concerning insulation used inside containment. It was noted that the Ginna plant does use insulation which could conceivably cause some blockage of the

sump intake screens. However, no specific review of the mechanism to cause such blockage has been made for the Ginna plant. It would require a combination of several low probability occurrences, such as a large major pipe break, loss of significant quantities of insulation due to resultant dynamic effects, and migration of this insulation to the sump screens to cause virtually complete blockage. RG&E is awaiting the draft proposed revision to Regulatory Guide 1.82, and the NRC's value-impact assessment as a result of plant-specific studies, in order to define whether any further analysis is required at Ginna.

Based on the extensive NPSH margin available for the RHR pumps at Ginna, and the preliminary results of the sump studies showing little effect on pump performance due to effects such as debris blockage and entrainment, RG&E considers that this issue is being adequately addressed for the Ginna plant, and that operation can continue without undue risk to the health and safety of the public.

Unresolved Safety Issue A-44

Station Blackout

Description of Problem:

The issue of station blackout involves a study of whether or not nuclear power plants should be designed to accommodate a complete loss of all A.C. power (i.e., a loss of off-site sources and all on-site emergency diesel sources). Loss of A.C. power for an extended time in pressurized water reactors, accompanied by loss of all of the auxiliary feedwater pumps, could result in a failure to adequately cool the reactor core, with potentially serious consequences.

Current NRC guidance states that, as a minimum, diverse power drives should be provided for the redundant auxiliary feedwater pumps. This is normally accomplished by utilizing one or more A.C. power electric motor driven pumps and one or more redundant steam turbine driven pumps, with the latter system operation relying only on D.C. power. One concern is the design adequacy of plants licensed prior to adoption of the current requirements.

RG&E Status:

The issue of A.C. power dependence of the auxiliary feedwater system was considered both as a TMI item, and in SEP Topic X. The Ginna design includes a 200% steam-driven AFW pump, as well as four 100% motor-driven AFW pumps. Regulatory reviews have

concluded that, based on system design and testing, the steam-driven auxiliary feedwater pump could provide its safety function without relying on A.C. power.

The issue of onsite and offsite power reliability was also extensively reviewed during the conduct of the SEP. It was concluded during the review of SEP Topics VIII-1.A, "Potential Equipment Failure Associated With a Degraded Grid Voltage" and VIII-2, "Onsite Emergency Power Systems - Diesel Generators" that the Ginna onsite and offsite A.C. power systems have very high reliability. Also, in the review of SEP Topic VIII-3.A, "Station Battery Capacity Test Requirements," it was concluded that the 8 hour capacity of the onsite batteries was sufficient to ensure adequate D.C. power to the station, until A.C. power could be restored in the event of simultaneous failures of both the on-site and offsite A.C. power systems. An Emergency Procedure, E-4.3, "Loss of A.C. Power" has been developed to detail the required actions.

RG&E thus concludes that the issue of Station Blackout is being adequately addressed for the Ginna Station, and that operation can continue without undue risk to the health and safety of the public.

Unresolved Safety Issue A-45

Shutdown Decay Heat Removal Requirements

Description of Problem:

Under normal operating conditions, power generated within a reactor is removed as steam to produce electricity via a turbine generator. Following a reactor shutdown, a reactor produces insufficient power to operate the turbine; however, the radioactive decay of fission products continues to produce heat (so-called "decay heat"). Therefore, when reactor shutdown occurs, other measures must be available to remove decay heat from the reactor. All light water reactors (LWRs) share two common decay heat removal functional requirements: (1) to provide a means of transferring decay heat from the reactor coolant system to an ultimate heat sink and (2) to maintain sufficient water inventory inside the reactor vessel to ensure adequate cooling of the reactor fuel. The reliability of a particular power plant to perform these functions depends on the frequency of initiating events that require or jeopardize decay heat removal operations and the probability that required systems will respond to remove the decay heat.

The principal means for removing the decay heat in a pressurized water reactor (PWR) under normal conditions immediately following reactor shutdown is through the steam generators using the auxiliary feedwater system. Following the TMI-2 accident,

the NRC required plants to make improvements to the auxiliary feedwater systems. However, the NRC staff further believes that providing an alternative means of decay heat removal could substantially increase the plants' capability to deal with a broader spectrum of transients and accidents and, therefore, could potentially significantly reduce the overall risk to the public. Consequently, this Unresolved Safety Issue will investigate alternative means of decay heat removal in PWR plants, including but not limited to using existing equipment where possible. This study will consist of a generic systems evaluation and will result in recommendations regarding (1) the adequacy of existing shutdown decay heat removal requirements, and (2) the desirability of and possible design requirements for an alternative decay heat removal method, that is, a method other than that normally associated with the steam generator and secondary system.

RG&E Status:

The design and qualification of the auxiliary feedwater system was reviewed both as part of the TMI review, and as part of SEP Topics X, "Auxiliary Feedwater Systems," and V-10.B, "RHR Reliability." The present auxiliary feedwater system consists of two 100% motor-driven auxiliary feedwater pumps, a 200% steam-driven auxiliary feedwater pump (independent of A.C. power), and two 100% motor-driven Standby Auxiliary Feedwater pumps. The motor driven pumps normally take suction from on-site Condensate Storage Tanks, but can also get water from the Service Water System (Lake Ontario). Furthermore, a modification made during

the SEP review of Ginna provided for connections allowing the use of the yard fire hydrant system (independent of on-site or off-site power) as a source of water from the motor-driven and steam-driven pumps. A similar modification is to be made for the Standby AFW pumps. It is thus apparent that many diverse means of water supply to and from the Auxiliary Feedwater Systems are available at Ginna. During the course of the Appendix R Fire Protection reviews, RG&E identified a means of going from hot shutdown to cold shutdown conditions could include filling of the steam generators and steam lines solid with water, and using them as a water-to-water heat exchanger. This method has been accepted as viable by the NRC.

Other means of removing decay heat have also been investigated, and described in the NRC's "Safe Shutdown Evaluation" for the Ginna SEP (September 29, 1981 report). These include use of the CVCS, RHR, steam generator blowdown systems, and the "bleed-and-feed" method, using the pressurizer PORV's and the Safety Injection pumps.

Given the extensive diversity and capacity of the Ginna auxiliary feedwater systems, and the other methods available for alternative decay heat removal means, RG&E considers that this issue is being adequately addressed for the Ginna plant, and that operation can continue without undue risk to the health and safety of the public.

Unresolved Safety Issue A-46

Seismic Qualification of Equipment in Operating Plants

Description of Problem:

The design criteria and methods for the seismic qualification of mechanical and electrical equipment in nuclear power plants have undergone significant change during the course of the commercial nuclear power program. Consequently, the margins of safety provided in existing equipment to resist seismically induced loads and perform intended safety functions may vary considerably among plants licensed in different time frames. The NRC staff has determined that the seismic qualification of the equipment in operating plants should be reassessed to ensure the ability to bring the plant to a safe shutdown condition when subject to a seismic event. The NRC's objective of this Unresolved Safety Issue A-46 is to establish explicit guidelines that can be used to judge the adequacy of the seismic qualification of mechanical and electrical equipment at all operating plants in lieu of attempting to backfit current design criteria. This guidance will concern equipment required to safely shut down the plant, as well as equipment whose function is not required for a safe shutdown, but whose failure could result in adverse conditions which might impair shutdown functions.

RG&E Status:

The Systematic Evaluation Program, through the Senior Seismic Review Team, performed an audit of the Ginna safety-related structures, systems, and components. This report is provided as NUREG/CR-1821. Most equipment was found to be capable of withstanding the Ginna SSE. In certain areas sufficient documentation was not available. Reanalysis and, in some cases, redesign or resupport are being conducted. The status of these items is provided in NUREG-0821.

In addition, the anchorage of major equipment was addressed. Experience from major earthquakes has shown that almost all seismically induced equipment failures in quality industrial facilities have occurred because the components were not adequately anchored to their foundations, and that few equipment failures have occurred in equipment that was anchored. As a result of the review of electrical equipment anchorage, modifications to upgrade the anchorages of a number of safety-related electrical components at Ginna were made.

RG&E is also participating in a Seismic Qualification Utility Group which is conducting a pilot program to explore an alternative method for seismically qualifying selected nuclear plant components based on experience with the equipment during earthquakes. This program is expected to assist the NRC and its consultants in developing qualification methodology for installed equipment at operating plants, in screening and assigning qualification priorities for more efficent utilization of NRC and industry resources,

and possibly in qualifying certain classes of equipment on a generic basis without specific testing or analyses of components.

Based on the above discussion, RG&E considers that this issue is being adequately addressed, and that operation of the Ginna plant can continue without undue risk to the health and safety of the public.

Unresolved Safety Issue A-47

Safety Implications of Control Systems

Description of Problem:

This issue concerns the potential for transients or accidents being made more severe as a result of the failure or malfunction of control systems. These failures or malfunctions may occur independently, or as a result of the accident or transient under consideration. One concern is the potential for a single failure (such as loss of power supply, short circuit, open circuit, or sensor failure) to cause simultaneous malfunction of several control systems. Another concern is for a postulated accident to cause control system failures which would make the accident more severe than analyzed. Accidents could conceivably cause control system failures by creating a harsh environment in the area of the control equipment or by physically damaging the control equipment.

Although it is generally believed that such control system failures would not lead to serious events or result in conditions that safety systems cannot safely handle, rigorous in-depth studies have not been performed to confirm this belief.

RG&E Status:

The separation of the Reactor Protection System from non-safety systems was the subject of SEP Topic VII-1.A. The SER for this

SEP topic concluded that the Ginna plant met all current licensing criteria.

Additional studies probing the interaction of safety and non-safety systems were performed during Ginna's fire protection reviews in response to 10 CFR50 Appendix R. Within designated fire zones, it was assumed that damage to any equipment (or its control cables, if affected) could cause failure of any type (including "hot shorts"). The dedicated shutdown system proposed by RG&E as a result of the fire protection study will incorporate the required separation of safety and non-safety systems.

In response to IE Information Notice 79-22, "Potential Unreviewed Safety Question on Interaction Between Non-Safety-Grade Systems and Safety-Grade Systems," RG&E performed an evaluation of these potential effects. By letter dated October 5, 1979, RG&E concluded that none of the scenarios constituted an unreviewed safety question for the Ginna plant.

Another potential control and safety system interaction was addressed in RG&E's response to the NRC's September 16, 1980 letter regarding loss of D.C. sources and inverters. RG&E's instrumentation bus and power supply arrangement is such that loss of any D.C. source or inverter would not result in the loss of any instrument buses. This is described in our letter of October 9, 1981.

Based on the interaction studies done to date, the separation already provided between safety and non-safety systems, and proposed fire protection modifications, RG&E considers that this issue is being adequately addressed, and that operation of Ginna can continue without undue risk to the health and safety of the public.

Unresolved Safety Issue A-48

Hydrogen Control Measures and Effects of Hydrogen Burns on Safety Equipment

Description of Problem:

Following a LOCA in an LWR plant, combustible gases, principally hydrogen, may accumulate inside the primary reactor containment as a result of: (1) metal-water reaction involving the fuel element cladding; (2) radiolytic decomposition of the water in the reactor core and the containment sump; (3) corrosion of certain construction materials by the spray solution; and (4) synergistic chemical, thermal, and radiolytic effects of post-accident environmental conditions on containment protective coating systems and electric cable insulation.

In the event of a severely degraded core, a large additional amount of hydrogen could be generated as a result of the reaction betweeen the molten fuel and the concrete containment base. Other combustible gases may also be generated by this reaction.

The accident at TMI-2 on March 29, 1979 resulted in metal-water reaction which involved hydrogen generation well in excess of the amounts specified in the current regulations 10 CFR Section 50.44. As a result, it became apparent to the NRC that additional hydrogen control and mitigation measures may need to be considered for all nuclear power plants.

RG&E Status:

The Ginna plant has redundant hydrogen recombiners, which would be used to prevent the accumulation of combustible mixture of hydrogen gas, in accordance with the guidance provided in Regulatory Guide 1.7. Also, the containment purge system includes filters, and could be used in the event of a hydrogen buildup. As a result of TMI modifications, redundant hydrogen monitors have been installed.

The great amount of hydrogen generated at TMI-2 was due to the failure to maintain adequate Safety Injection flow. Based on intensive training of operators, as well as procedural modifications, it is not expected that this would re-occur. Further, it has been estimated, during the reviews of the Zion and Sequoyah containments, that a reactor containment building could withstand pressures 2 to 3 times design pressure for short periods of time without catastrophic failure, and that a large dry reinforced concrete containment such as that used at Ginna is least susceptible to hydrogen burn considerations because of the large volume and high design pressure.

Because of the hydrogen monitoring and control systems in place at Ginna, the extensive procedural and training upgrades as a result of the accident at TMI-2, and the pressure margins available at Ginna-type containments, RG&E considers that the hydrogen burn issue is being adequately addressed, and that operation of Ginna can continue without undue risk to the health and safety of the public.

Unresolved Safety Issue A-49

Pressurized Thermal Shock

Description of Problem:

As a result of operating experience, it is recognized that transients can occur in pressurized water reactors characterized by severe overcooling causing thermal shock to the vessel, concurrent with or followed by repressurization. In these pressurized thermal shock (PTS) transients, rapid cooling of the reactor vessel internal surface causes a temperature distribution across the reactor vessel wall. This temperature distribution results in thermal stress with a maximum tensile stress at the inside surface of the vessel. The magnitude of the thermal stress depends on the temperature differences across the reactor vessel wall. Effects of this thermal stress are compounded by pressure stresses if the vessel is repressurized.

Severe reactor system overcooling events which could be followed by repressurization of the reactor vessel (PTS events) can result from a variety of causes. These include instrumentation and control system malfunctions, and postulated accidents such as small break loss-of-coolant accidents (LOCAs), main steamline breaks (MSLBs), feedwater pipe breaks, or stuck open valves in either the primary or secondary system.

As long as the fracture resistance of the reactor vessel material remains relatively high, such events are not expected to cause failure. After the fracture toughness of the vessel is reduced by neutron irradiation (and this occurs at a faster rate in vessels fabricated of materials which are relatively sensitive to neutron irradiation damage), severe PTS events could cause crack propagation of fairly small flaws that are conservatively postulated to exist near the inner surface.

RG&E Status:

Rochester Gas and Electric is an active participant in the Westinghouse Owners Group, which is evaluating the PTS questions for Westinghouse reactors. Analyses performed to date indicate that the Ginna reactor vessel could withstand a severe overcooling event.

Since analyses have demonstrated the capability of the Ginna reactor vessel to withstand severe overcooling transients, and RG&E is continuing with the Westinghouse Owners Group and the NRC to reach a final solution to the PTS issue, RG&E considers that this issue is being adequately addressed, and that the Ginna plant can continue to operate without undue risk to the health and safety of the public.