U.S. NUCLEAR REGULATORY COMMISSION
OFFICE OF INSPECTION AND ENFORCEMENT

Region I

Report No. 50-244/79-13

Docket No. 50-244

License No. DPR-18 Priority -- Category C

Licensee: Rochester Gas and Electric Corporation
89 East Avenue
Rochester, New York 14649

Facility Name: R. E. Ginna Nuclear Power Plant, Unit 1

Inspection at: Ontario, New York

Inspection conducted: September 17-20, 1979

Inspectors: [signature] for 10/24/79
R. S. Mankowski, Reactor Inspector date signed

date signed

date signed

Approved by: [signature] 10-24-79
R. R. Keimig, Chief, Reactor Projects Section No. 1, RO&NS Branch date signed

Inspection Summary:

Inspection; September 17-20, 1979 (Report No. 50-244/79-13):
Areas Inspected: Routine, unannounced inspection which included the review of: plant operations; activities of the onsite review committee (PORC); nonrountine event reporting program; procedure changes associated with IE Bulletin 79-06C, item 1; potential failure mechanism of the pressurizer power operated relief valve yoke; and initial site review of support deficiences identified in the "D" Standby Auxiliary Feedwater System. The inspection involved 31 inspector-hours onsite by one NRC regional based inspector.
Results: No items of noncompliance were identified.

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(Rev. April 77)

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DETAILS

1. Persons Contacted

   *Mr. W. Backus, Operations Supervisor
   Mr. J. Bodine, QC Inspection Engineer
   Mr. G. Larriza, Technical Engineer
   Mr. M. Lilley, Mechanical Engineer
   Mr. J. Noon, Assistant Superintendent
   *Mr. C. Peck, Operations Engineer
   *Mr. T. Schuler, QC Engineer
   *Mr. B. Snow, Superintendent

   The inspector also interviewed other licensee personnel including members of the operations, engineering, maintenance, health physics, general office and quality assurance staffs.

   *Denotes those present at the exit interview.

2. Plant Operations Review

   a. Shift Logs and Operating Records

      (1) The inspector reviewed the following logs and records:

         -- Shift Foreman Record, September 1-14;

         -- Head Control Room Operators Logs, September 1-14; and,

         -- Auxiliary Operators' Primary Side Logs, September 1-19.

      (2) The logs and records were reviewed to verify that:

         -- Log sheet entries are filled out and initialed;

         -- Log entries involving abnormal conditions are sufficiently detailed;

         -- Log book reviews are being conducted by the staff; and,

         -- Problem identification reports confirm compliance with TS reporting and LCO requirements.

      No items of noncompliance were identified.

b. Plant Tour

The inspector conducted a tour of the Auxiliary Building and the Control Room. During this tour, the inspector observed the following:

-- local and remote valve positions indication and breaker alignment associated with the Boric Acid Storage Tanks, Component Cooling Water, Safety Injection System, NaOH Injection Tank;

-- security measures associated with the new fuel storage area;

-- general cleanliness and general control of anti-contamination clothing;

-- general control of Radiation Areas;

-- Boric Acid Storage Tank, Refueling Water Storage Tank; NaOH Spray Additive Tank water volumes; and

-- switch alignment associated with the Reactor Protection System Nuclear Instrumentation Channels.

Additionally, the control board was observed for annunciators that should not normally be lighted during the existing plant conditions. The reason for each alarmed annunciator was adequately explained by the Shift Foreman.

Acceptance criteria for the above review included inspector judgement and requirements of applicable Technical Specifications.

No items of noncompliance were identified.

3. Review of Activities of the Plant Operations Review Committee (PORC)

a. References

-- Technical Specifications, Appendix A, Section 6.5.1; and,

-- A-17, Plant Operations Review Committee Operating Procedure, Revision 26.

b. Review

The inspector reviewed procedure A-17 to verify that it implemented the requirements of Technical Specification 6.5.1.

The inspector also selected and reviewed the PORC minutes listed below to verify that documentary evidence indicated that the requirements of the procedure were being complied with.

The minutes reviewed were:

-- 78-48 through 79-60, June 12, 1978 - June 29, 1979 (to establish meeting frequency requirements were met);

-- 78-62 through 78-66, August 21-28, 1978;

-- 78-85 through 78-88, November 15-December 4, 1978; and,

-- 79-05 through 79-08, January, 1979.

No items of noncompliance were identified.

4. Non Routine Event Reporting Program

a. References

-- A-17, Plant Operations Review Committee Operating Procedure, Revision 26;

-- A-25, Reporting of Unusual Plant Conditions, Revision 7;

-- A-25.1, Ginna Station Event Report; Revision 11;

-- A-52.4, Control of Limiting Conditions for Operating Equipment, Revision 19;

-- A-52.5, Control of Limiting Conditions for System Specifications, Revision 4;

-- A-1601, Corrective Action Report, Revision 1; and,

-- A-1602, Preventive Action Report, Revision 1.

b. Program Review

The inspector reviewed the above referenced procedures to verify that administrative controls have been established for the following:

-- prompt review and evaluation of off-normal operating events;

-- prompt review of deficiencies identified during equipment maintenance and testing;

-- reporting off-normal operating events internally and to the NRC;

-- completion and review of corrective action relating to off-normal operating events;

-- assignment of responsibility for the identification, disposition and review of action taken to correct conditions causing off-normal operating events; and,

-- formal review of Westinghouse Technical Bulletins for plant applicability.

No items of noncompliance were identified.

5. IE Bulletin 79-06C, Item 1

a. References

-- A-52.1, Shift Organization Relief and Turnover, Revision 6;

-- E-1.1, Safety Injection System Actuation, Revision 11;

-- E-1.2, Loss of Coolant Accident, Revision 16;

-- E-1.3, Steam Line Break Accident, Revision 5;

-- E-1.4, Steam Generator Tube Rupture Accident, Revision 5; and,

-- E-15.1, Malfunction of Power Relief or Safety Valves, Revision 4.

b. Review

By letter dated August 29, 1979, in response to IEB 79-06C, item 1, the licensee stated that licensed operators had been instructed to trip all Reactor Coolant Pumps upon reactor trip and initiation of safety injection at a Reactor Coolant System pressure of 1715 psig; and, that two licensed operators will be stationed in the control room at all times during plant operations.

The inspector reviewed the above referenced procedures and confirmed that they had been revised and were consistent with the licensee's response except as discussed below.

During the initial review of E-1.4, the inspector noted that step 2.10 required that the Reactor Coolant Pumps be secured at 1500 psig vice 1750 psig as stated in the bulletin response. The licensee immediately convened a Plant Operations Review Committee meeting, formally reviewed and approved a change to E-1.4 incorporating the 1750 psig provision, distributed the revised procedure to the control room and provided documented notice to the licensed operators, via the Operation Plans, that E-1.4 had been revised.

No items of noncompliance were identified. However, the other actions required to be performed by the bulletin will be reviewed during a subsequent inspection after formal review by the NRC of the Westinghouse Owners Group procedure guidelines, submitted as Section 6 and Appendix A of WCAP 9600 (79-BU-6C).

6. Review of PORV Discrepancies

During an NRC inspection conducted at the Westinghouse Water Reactor Division Plant on September 13, 1979, a malfunction in a Pressurizer Power Operated Relief Valve (PORV) which had occurred at a Swiss reactor plant in 1974 was identified. The malfunction occurred during power operation and resulted in an inadvertent opening of the valve. The resulting transient was terminated by closure of the PORV block valve by the operator.

The subject valve was supplied by Copes Vulcan under Westinghouse Specification 6676270. The valve malfunctioned due to failure of the cast iron valve yoke. The review conducted during the inspection at Westinghouse indicated that a similar valve(s) may be installed at the R. E. Ginna Nuclear Power Plant.

The licensee was notified by telephone from the Region I office on September 17, 1979. The inspector, in conjuction with the licensee, reviewed the valve technical manual and drawings and determined that the PORVs' installed at Ginna Station were supplied under Westinghouse Specification 676270 (apparently, the same specification) and the yoke material was "ductile iron". Further information received from the Region 1 office indicated that "cast steel" had been preliminarily determined to be the only acceptable yoke material.

The inspector reviewed the procedure (E-15.1) for inadvertent opening of the PORV, reviewed the Operations Plan and discussed the PORV valve position indication with Operation's supervisory personnel. The inspector determined that:

-- E-15.1, Malfunction of Power Operated Relief Valve or Safety Valves, Revision 4, identified those plant parameters which would indicate PORV opening and required closure of the appropriate block valve;

-- the Operations Plan as of September 17, 1979, documented the potential failure mode and required licensed operators to review and sign the entry; and,

-- PORV position indication is provided in the control room (status lights) and is operated by stem mounted limit switches.

On September 19, 1979, a letter was transmitted to the licensee by the NRC requesting information, pursuant to 10 CFR 50.54 (f), concerning the PORVs' installed. Pending response to this request for information, this item is unresolved. (79-SP-01)

7. Pipe Support Inadequacies

On September 21, 1979, the inspector was informed that support deficiencies were identified in the "D" Standby Auxiliary Feedwater System Service Water suction lines. The supports involved were identified as SW-140, 141, 142, 143 and 145. They consisted of structural steel attached to a concrete surface with anchor bolts (Hilti·Kwik.Bolts) without the use of base plates. The deficiencies consisted of loose nuts and gaps between the steel and the concrete surface. The licensee stated that they had not considered this type of support during the review and evaluation required by IEB 79-02, "Pipe Support Base Plate Designs Using Concrete Expansion Anchor Bolts".

The inspector toured the Standby Auxiliary Feedwater Pump Room and, on a sampling basis, visually inspected supports in the "C" Standby Auxiliary Feed System. No discrepancies were visually detected during this tour.

Subsequent to the inspection, further review by NRC management indicated that this support type, i.e., the attachment of a structural steel member to a concrete surface with anchor bolts without the use of a "baseplate" was intended to be included for evaluation under IEB 79-02.

By telephone, on October 1, 1979 the Plant Superintendent was notified by the inspector of this position and was requested to determine promptly the extent to which this type of support was used in Phase 1 systems (both accessible and inaccessible), to provide detailed plans for the evaluation of the adequacy of these supports and to notify the NRC Region I office of the results of the review and schedule.

Pending receipt of this information and reinspection, this item is unresolved (79-BU-02).

8. Unresolved Items

Unresolved items are matters about which more information is required to determine whether they are acceptable, items of noncompliance or deviations. Unresolved items are discussed in paragraph 6 and 7.

9. Exit Meeting

A management meeting was held with licensee representatives at the close of the inspection on September 20, 1979. A subsequent telephone call was held with the Plant Superintendent on October 1, 1979. The scope and findings as discussed in this report were presented at the meeting and during the subsequent telephone call.