U. S. NUCLEAR REGULATORY COMMISSION
REGION I

Report No. 83-11

Docket No. 50-244

License No. DPR-18 Priority -- Category C

Licensee: Rochester Gas and Electric Corporation

49 East Avenue

Rochester, New York 14649

Facility Name: R. E. Ginna Nuclear Power Plant

Inspection At: Rochester, NY, Ontario, NY and Reading, Pa.

Inspection Conducted: May 2-6, 1983

Inspectors: W. H. Baunack 6/7/83
W. H. Baunack, Project Engineer date

P. K. Eapen 6/7/83
P. K. Eapen, Reactor Engineer date

Approved by: HB Kister 6/7/83
H. B. Kister, Chief, Reactor Projects Section 2C, Division of Project and Resident Programs date

Inspection Summary: Inspection on May 2-6, 1983 (Report No. 50-244/83-11)
Areas Inspected: Special safety inspection by two region-based inspectors (49 hours) of licensee actions taken to comply with requirements described in NUREG-0737, Item II.B.2, Design Review of Plant Shielding.

Results: No violations were identified.

DETAILS

1. Persons Contacted

Rochester Gas and Electric

G. Larizza, Operations Engineer
C. Manbretti, Nuclear Engineer
*R. Mecredy, Manager Nuclear Engineering
D. Travis, Operations Aide

Gilbert/Commonwealth Engineers and Consultants

R. Anderson, Project Engineer
H. Manning, Manager Corporate QA Program
C. Paschall, Manager Design Control
M. Waselus, Section Manager, Nuclear Analysis

*present at exit interview on May 5, 1983.

The inspectors also interviewed and talked with other licensee personnel during the course of the inspection.

2. Plant Shielding Design Review

a. Background and Scope

As discussed in Item II.B.2 of NUREG-0737, "Clarification of TMI Action Plan Requirements," each power reactor licensee was required to perform a radiation and shielding design review of spaces around systems that may, as a result of an accident, contain highly radioactive materials. The design review was intended to identify the location of vital areas* and equipment in which personnel occupancy may be unduly limited or safety equipment may be unduly degraded by the radiation fields during post-accident operations of these systems. Additionally, each licensee was required to provide for adequate access to vital areas and protection of safety equipment by design changes, increased permanent or temporary shielding, or post-accident procedural controls. The design review was to determine which types of corrective actions were needed for vital areas throughout the facility.

These requirements were discussed in Item 2.1.6.b of NUREG-0578, "TMI-2 Lessons Learned Task Force Status Report and Short-Term Requirements"; were issued by NRC letters dated September 13 and October 30, 1979 to all operating nuclear power plants; and were incorporated into NUREG-0660, "TMI-2 Action Plan." Significant changes in requirements or guidance were described in an NRC letter

*Any area which will or may require occupancy to permit an operator to aid in the mitigation of or recovery from an accident is designated as a vital area.

to all licensees of operating plants dated September 5, 1980, and were subsequently described in Item II.B.2 of NUREG-0737. Lastly, an NRC letter to all licensees of operating power reactors dated March 17, 1982 (Generic Letter No. 82-05) requested reconfirmation of the schedule for completing Item II.B.2 of NUREG-0737.

With respect to operating power reactor licensee's, the October 30, 1979 NRC letter indicated that licensee's plant shielding design reviews were among those items for which post-implementation NRC review is acceptable. Although prior NRC approval was not required, licensees were to document their methods of implementation by the required completion date, e.g., design review by January 1, 1980 and plant modifications by January 1, 1981.

With respect to documentation specified by NUREG-0737 for vital area access, operating license applaicants were to provide to the NRC a summary of the shielding design review, a description of the results of this review, and a description of the modifications made or to be made to implement the results of the review. The submittals were to include:

(1) Specification of source terms used in the evaluation, including time after shutdown that was assumed for source terms in systems;

(2) Specification of systems assumed in the analysis to contain high levels of radioactivity in a post-accident situation;

(3) Specification of areas where access is considered necessary for vital system operation after an accident; and,

(4) The projected doses to individuals for necessary occupancy times in vital areas and a dose rate map for potentially occupied areas.

NUREG-0737 did not state that licensees of operating reactors were to submit the above documentation to the NRC. Rather, they were to have available for review the final design details of the implementation* of the Item II.B.2 position and clarification. (Information equivalent to that submitted by operating license applicants is expected to be available for review as documentation of the design review that provided the bases for final design details.) If deviations to that position and clarification were necessary, licensees were to provide detailed explanation and justification for the deviations by January 1, 1981.

The licensee's plant shielding design review and corrective actions were reviewed during this inspection. This review included licensee submittals to the NRC, a sampling verification of the shielding design review methodology and representative calculations, a review of selected emergency procedures to determine if the vital areas where personnel must go are safely accessible, and a review of corrective actions taken or planned by the licensee (including plant modifications).

*In addition to providing clarification of requirements, NUREG-0737 revised the completion date for modifications resulting from the plant shielding design review to January 1, 1982.

b. Licensee Submittals To The NRC and Previous Staff Evaluation

In the case of the R. E. Ginna Nuclear Power Plant, the shielding Design Review and Planned Corrective Actions were discussed by Rochester Gas and Electric Corporation (the licensee) in letters to the NRC dated October 17, 1979, November 19, 1979, and December 28, 1979. The licensee's shielding study and planned actions were evaluated by the NRC staff to meet the Category "A" lessons learned requirements for this item (NUREG-0578 Item 2.1.6.b) as discussed in an NRC letter to the licensee dated July 7, 1980. The licensee subsequently discussed the status of required actions including a summary description of the design changes and the implementation schedule for modifications, in letters to the NRC dated December 15, 1980, September 4, 1981, November 25, 1981, September 16, 1982, April 23, 1982, and December 2, 1982.

The above licensee and NRC letters were reviewed during this inspection to determine the licensee action completed or to be taken, and the extent of previous staff evaluation, regarding the plant shielding design review for the Ginna Plant. The licensee's statements and commitments contained in these letters, provided the bases, in part, for the inspectors verification that plant shielding modifications have been adequately identified and implemented (or scheduled for implementation), as discussed in other paragraphs of this report.

c. Shielding Design Calculation and Dose Estimates

The inspector reviewed the details of the licensee's Shielding design review calculations with the licensee's contractor (Gilbert/Commonwealth Engineers and Contractors) representatives. These details included the mathematical models, assumptions, source terms, dose rates and doses to personnel during post accident access to vital areas. The licensee's contractor obtained the source terms, dose rates and cummulative doses from a Standard Westinghouse Reactor. The plant specific adjustments to the source terms and isotopic concentrations were done using an inhouse computer code namely RWDS Code. The gamma radiation doses were estimated using another inhouse computer code (SDC Computer Code). The instantaneous doses were integrated using yet another in house code (LSQ_2) to obtain cummulative doses.

The inspector reviewed the mathematical models and bench marking for the above computer programs and found these to be adequate. The documents indicated that the mathematical models used were consistent with the State-of-the-Art methods for shielding calculations. The inspector also compared the results of the contractor's dose calculations with those from the NRC consultant's calculations for similar configurations and results were consistent.

The inspector also reviewed the contractor's methods for determining the doses to personnel from post accident access to vital areas. There methods and the estimated doses are reasonable and acceptable.

The inspector has no further questions in this regard.

d. Assessment of Vital Areas and Corrective Actions

The inspector reviewed the licensees contractors assessment of vital areas, as decribed in a December, 1979 Report Titled" Design Review of Plant Shielding and Environmental Qualification of Equipment for Spaces/Systems which may be used in Post Accident Operations Outside Containment at R. E. Ginna Nuclear Power Plant". This review was accomplished using the recommendations and guidelines of NUREG-0578. This evaluation identified the most critical areas requiring personnel access following the onset of extreme accident conditions following a postulated major release of radioactivity into the containment building. Consideration was given to areas where predetermined post accident functions would be performed (Nuclear Sample Room, Chemistry Laboratory, Count Room, Control Room and air sample penetrations) and to areas where personnel could be called upon to execute certain accident-mitigating or short-term recovery tasks (Hydrogen recombiner panel, radwaste panel and building ventilation filters). Potential radiation exposures were determined for each task and included additional exposure due to accessing the areas considered.

The licensees further review and evaluation identified certain modifications which were necessary to lower dose rates for certain tasks which must be performed to aid in the mitigation of or recovery from an accident. These modifications included (1) count room wall shielding (2) floor reinforcing to support the count room wall shielding (3) verifical shield wall extension at the sample line containment penetration area (4) shield wall end closure door at the sample line containment penetration area (5) waste gas vent header shielding and relocation (6) installation of a post-accident sampling system, and (7) installation of a new radwaste control panel.

e. Vital Area Accessibility Procedure Review

The inspector reviewed selected procedures that would be implemented by the licensee in the event of a loss of coolant accident. The review included (1) a plant walkdown of the procedures to determine the ability to perform the procedural steps and the accessibility of manual valves or breakers that may require local operation, and (2) an assessment of potential exposures to plant personnel based on the results of the licensee's shielding design review. The procedures reviewed included Procedure No. E-1.2, Loss of Reactor Coolant, Revision 28, Dated October 14, 1982; Procedure No. E-1.5, Void Formation in the RCS, Revision 11, dated August 4, 1982; Procedure No. E-1.1, Immediate Action and Diagnostics for Spurious Actuation of

SI, LOCA, Loss of Secondary Coolant, and Steam Generator Tube Rupture, Revision 26, dated October 19, 1982; and, Procedure No. 0-8.1, Restoration of Pressurizer heaters to maintain natural circulation at HSD, Revision 2, dated July 7, 1982.

Based on this review it appears the procedures currently require personnel entry into areas in which, under certain postulated plant conditions, high radiation dose rates may be encountered. The licensee stated that a review of procedures to determine vital areas which must be accessed following an accident has not been performed. But, that such a review will be conducted and that this review will consider the potentially radioactive lines which have been installed for the new post accident sampling system. The results of the licensee's review will be evaluated during a future inspection. (244/83-11-01)

3. Implementation of Plant Shielding Modifications

The inspectors reviewed the licensee's implementation of modifications that resulted from the plant shielding design review discussed in paragraph 2. The inspectors reviewed the Safety Evaluations, Design Inputs, and Design Criteria for the modifications listed below:

Engineering Work Request (EWR) 2828, TMI Shielding Modifications.

- Increased height of shield wall across from nuclear sample room.
- Installed lead door on penetration shield.
- Relocated vent header line in auxiliary building intermediate level.
- Shielded portion of vent header line in auxiliary building intermediate level.
- Installed lead shielding on east wall of count room.
- Reinforced count room floor.

EWR 3037, Radwaste Computer Control System.

- Installation of a radwaste control panel.

EWR 2606, Post Accident Sampling System Implementation.

- Installation of a post accident sampling system.

With respect to the plant modifications approved by the above engineering design documents, the inspectors verified that the modifications had been properly reviewed, approved, and controlled. The inspectors observed the completed modifications associated with EWR 2828 and the status of completion of EWR's 3037 and 2606 during a plant tour on May 5, 1983.

The inspectors had no further questions relating to plant shielding modifications.

4. Unresolved Items

Unresolved items are matters about which more information is required in order to ascertain whether they are acceptable, an item of noncompliance, or a deviation. An unresolved item is discussed in paragraph 2.e.

5. Exit Interview

The inspectors met with the licensee representative (denoted in paragraph 1) at the conclusion of the inspection on May 5, 1983, to discuss the inspection scope and findings, as described in this report.