50-244

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MEMORANDUM FOR: James C. Linville, Chief
Reactor Projects Branch #3
Division of Reactor Projects
Region I

FROM: Jack E. Rosenthal, Chief
Reactor Operations Analysis Branch
Division of Safety Programs
Office for Analysis and Evaluation
of Operational Data

SUBJECT: EVALUATION OF LERs FOR R. E. GINNA NUCLEAR STATION FOR SALP INPUT FROM OCTOBER 1, 1990, TO JANUARY 18, 1992

In support of the ongoing SALP reviews, AEOD has reviewed the licensee event reports (LERs) submitted by Rochester Gas and Electric Corporation for R. E. Ginna Nuclear Power Plant. Our review concentrated on the safety significance of the events, LER completeness, clarity, understandability and adequacy of the event report contents.

The enclosure provides additional observations from our review of the LERs. If you have any questions regarding this report, please contact either myself or Sal Salah of my staff. Sal can be reached at FTS 492-4432.

Original signed by
Jack E. Rosenthal, Chief
Reactor Operations Analysis Branch
Division of Safety Programs
Office for Analysis and Evaluation
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Enclosure: As stated
cc w/enclosure:
A. Johnson, NRR
A. Mendiola, NRR
T. Moslak, SRI R. E. Ginna

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ENCLOSURE

AEOD INPUT TO SALP REVIEW FOR R. E. GINNA NUCLEAR POWER PLANT

Rochester Gas and Electric Corporation submitted about 16 reports for R. E. Ginna Nuclear Power Plant, not including updates, in the assessment period from October 1, 1990, to January 18, 1992. Our review included the following LER numbers:

90-013 to 90-019
91-001 to 91-009

The LER review followed the general instructions and procedures of NUREG-1022. Our review findings are given below:

1. Important Operating Events

There were three reported events at R. E. Ginna plant that were identified as important events by AEOD screening and review process in the assessment period. These events were reported in LERs 90-013, 91-002 and 91-005.

LER 244/90-013: "Turbine Trip Due to an Inadvertent ATWS Mitigation System Actuation Circuitry (AMSAC) Actuation Causes a Reactor Trip." On December 11, 1990, with the reactor at approximately 97 percent full power, a turbine trip with subsequent reactor trip occurred due to an inadvertent ATWS Mitigation System Actuation Circuitry (AMSAC) actuation. The control room operators verified the reactor and turbine trips and performed the appropriate actions of emergency procedures E-O (Reactor Trips or Safety Injection) and ES-0.1 (Reactor Trip Response). The plant was subsequently stabilized in the hot shutdown condition.

The immediate cause of the AMSAC actuated turbine trip was determined to be due to a low voltage potential of a logic output from a Foxboro N-2CCA-DF control module in the AMSAC logic circuitry. This event was important because it represented vendor circuit design deficiency and several equipment failures occurred.

LER 244/91-002: "Loss of Offsite Power Circuit 751, Due to Ice Storm, Causes Automatic Actuation of ESF and RPS." On March 4, 1991, with the reactor at approximately 97 percent of full power and again on March 7, 1991, with the reactor at approximately 80 percent of full power, a turbine runback occurred and the "A" emergency diesel generator (EDG) started and tied into safeguard buses 14 and 18.

The control room operators performed the appropriate actions of automatic turbine runback and verified that the "A" EDG was operating properly and that safeguards buses 14 and 18 were energized. In each of these events, the plant was subsequently stabilized at approximately 20 percent less power than when the event started. The cause of the events was the loss of power to safeguards buses 14 and 18 due to loss of offsite power from

34.5 kV circuit 751. Circuit 751 was lost due to RG&E system disturbances caused by a major ice storm.

Corrective action was to restore normal power to safeguards buses 14 and 18 from offsite power and to stop the "A" EDG and realign it for auto standby.

LER 244/91-005: "Steam Generator Tube Degradation Due to IGA/SSC Causes Q. A. Manual Reportable Limit to be Reached." During the 1991 annual refueling and maintenance outage subsequent to the eddy current examination performed on both the "A" and "B" Westinghouse Series 44 Steam Generator (S/G), the licensee found 166 tubes in the "A" S/G and 177 tubes in the "B" S/G required corrective action due to tube degradation.

The underlying cause of the tube degradation was due to a common S/G problem of a partially rolled tube sheet crevice with recurring intergranular attach/stress corrosion cracking (IGA/SCC) and primary water stress corrosion cracking (PWSCC) attack on S/G tubing.

2. PNs Issued in Assessment Period

No Preliminary Notification of Event or Unusual Occurrence reports were issued during the assessment period for R. E. Ginna Nuclear Station.

3. LER Quality

The LERs adequately described all the major aspects of the events, including all component or system failures that contributed to the events and the significant corrective actions taken or planned to prevent recurrence.

4. 10 CFR 50.72 Reports

Based upon our review of preliminary information provided by the licensee in immediate notification reports submitted pursuant to 10 CFR 50.72, it appears that all related LERs required by 10 CFR 50.73 were submitted to the NRC.