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MAIER,J.E. Rochester Gas & Electric Corp.
RECIP.NAME RECIPIENT AFFILIATION
CRUTCHFIELD,D. Operating Reactors Branch 5

SUBJECT: Forwards response to findings listed in Section 1.4 of NUREG-0909, "NRC Task Force Rept on 820125 Steam Generator Tube Rupture at Ginna Nuclear Power Plant," per NRC 820416 ltr.

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JOHN E. MAIER
Vice President

TELEPHONE
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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: Response to NUREG-0909
Steam Generator Tube Rupture Incident
R. E. Ginna Nuclear Power Plant
Docket No. 50-244

Dear Mr. Crutchfield:

By letter dated April 16, 1982, you transmitted to us NUREG-0909, "NRC Task Force Report on the January 25, 1982 Steam Generator Tube Rupture at the R. E. Ginna Nuclear Power Plant." You requested that we address each of the findings identified in the report or to identify where findings had previously been addressed. Attachment A to this letter responds to each of the findings listed in Section 1.4 of NUREG-0909.

Very truly yours,

JEMaier

J. E. Maier

Attachment

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Attachment A
Response to Section 1.4, Significant Findings
of NUREG-0909

1.4 Significant Findings

1.4.1 Facility Response

1. The steam generator tube failed suddenly and without any prior indication of a leaking tube.

Response: This statement is correct. During normal operation, two methods are used which could detect and quantify primary to secondary leakage: steam generator blowdown activity (I-131) and air ejector off gas noble gas activity (Xe-133). Assuming operating conditions such as those during 1981 and steady state operation, the blowdown sample, which is taken three times per week during normal operation, can detect 1 to 2 cc/minute primary to secondary leakage (3 to 5 x 10^{-4} gallons per minute). The air ejector off gas is normally checked at least monthly and would indicate .001 gallons per minute.

2. The initial reactor coolant system depressurization, caused by the break flow, probably resulted in some steam formation in the region under the relatively hot reactor vessel upper head....

Response: This finding is correct.

3. All engineered safeguard features operated as designed.

Response: This finding is correct.

4. After the reactor coolant pumps were tripped, natural circulation flow developed in both reactor coolant system loops....

Response: This finding is correct.

5. The reactor coolant system depressurization, resulting from manual operation of one pressurizer PORV resulted in steam formation in the relatively hot and stagnant areas of reactor vessel upper head region and the primary side of the top of the tube bundle of the faulted steam generator ...

Response: There is no absolute method to determine if steam voids formed in the faulted steam generator's tubes when the PORV was opened and the RCS depressurized to 835 psig. Since the normal no load saturation condition of the steam generator is approximately

1000 psig, it is possible that the primary fluid, which was relatively stagnant in the B steam generator, may have flashed as the pressure was reduced below 1000 psig. The actual pressure in the steam generator at the time of the PORV opening was approximately 960 psig. Fluid being introduced to the steam generator through the break was cooler than normal hot zero power. Some stratification of water in the steam generator may have existed with the tube bundle region cooler than the steam space. If that is true, the saturation pressure at the tube bundle was less than 960 psig. It is not possible to make a determination of void formation without knowing the temperature profile in the steam generator. Other statements in this finding are true.

6. One pressurizer PORV stuck open after three successful cycles and the corresponding block valve had to be shut....

Response: We agree with the first paragraph. The second paragraph indicates that increased restriction is required for the malfunction. This is not the only potential cause. The other possible sources are identified in Section 5.2.4 of Reference 1.

7. The main steam system valve position recorders failed to indicate openings of the faulted steam generator safety valve and both A and B steam generator atmospheric PORVs....

Response: The failure of the main steam safety valve position indication is discussed in Section 5.6 of Reference 1.

8. Based on Task Force analysis of available data from the instrumentation installed at Ginna, significant thermal shock of the reactor pressure vessel wall has not been ruled out....

Response: Using worst case conditions, analyses have demonstrated that the January 25, 1982 transient did not impair the integrity of the reactor coolant system. The analyses are discussed in Section 6.4 of Reference 1 and in Reference 2.

9. Two hours after the start of the event, a reactor coolant pump was restarted in the nonfaulted reactor coolant system loop, which equalized temperatures in the loops and reactor vessel, collapsed any remaining steam bubble in the reactor vessel upper head and faulted steam generator tube bundle areas, and assisted in stabilizing and cooling the reactor coolant system.

Response: This finding is correct.

10. The pressurizer relief tank rupture disc burst as a result of the tank being flooded....

Response: This statement is correct.

11. The lowest setpoint ASME Code safety valve on the faulted steam generator opened five times during the event....

Response: This finding is only partially supported by the evidence. The NRC water balance, provided in summary form in Sections 3.6 and 5.3.4 of NUREG-0909, is in error in the following aspects.

We have reviewed the NRC System Mass Balance calculation and have determined that there are two significant differences as compared to the RG&E mass balance calculations (Reference 1 Section 6.9) that lead to the larger inferred release of fluid through B-steam line safety valve by the NRC. In item 7 of Table 3.5, the NRC calculates feedwater addition to the B-Steam Generator based upon 17 minutes of operation for the turbine-driven auxiliary feedwater pump (AFWP). Discussion with plant personnel confirms that feedwater for both the motor-driven and turbine-driven AFWP was stopped after three minutes of operation. Secondly, the NRC accounts for steam loss prior to main steam isolation valve closure by subtracting 50% of feedwater mass addition. The origin of this 50% factor is not explained. Third, it appears that steam dump to condenser was not included. Based upon the computer alarm printout, steam dump valve open times and flow rates have been estimated resulting in a mass loss from the B-Steam Generator of 47,000 lbm.

The NRC estimated loss through the safety valve at 117,000 lbm. Subtracting from this the 2,000 gal. (17,000 lbm @ 60^{o}F) of the NRC feedwater addition (which included the steam loss), adding our total of 14,000 lbm from feedwater and subtracting steam dump losses of 47,000 lbm result in an adjusted NRC total for safety valve losses of 67,000 lbm.

Our review of the NRC calculation concludes that we have no basis to change our water balance calculation provided in Section 6.9 of Reference 1.

RG&E does not believe that the data can conclusively support the safety valve leaked after the first lift. Our evaluation is provided in Section 7.2 3 of Incident Evaluation dated April 12, 1982.

The 50 minute period referenced is from the completion of the fifth safety valve lift at 11:37 to 12:27. It is possible that the main steam line filled with water at 12:27 causing the B steam generator pressure to cross the reactor coolant loop pressure. It cannot be verified that the safety valve reseated, however, Section 7.2.3 of the Incident Evaluation Report supports the belief that it did.

Finally, the B-steam line pipe hangers outside containment were pinned due to a concern with dead weight load but not due to a water hammer concern.

12. Operators are expected to use the best means available to them to safely cope with an event, regardless of the safety classification of the equipment utilized....

Response: This finding is generally correct. We note that the loop temperature indication provided by the RTDs is safety related and complements the incore thermocouples. We also note that a fully qualified wide range loop pressure transmitter will be installed and available prior to startup.

The cause of the process computer failure is not known. At the time of failure, it was deemed more important to restart the computer than to investigate the cause.

13. Accurate analyses of the reactor coolant system water inventory, safety injection and charging flows, tube rupture flows, and safety valve releases could not be made with the permanent records generated at Ginna....

Response: We note the general agreement between NRC results, when revised as noted in our response to finding 11 above, and RG&E results as provided in Section 6.9 of Reference 1.

1.4.2 Human Factors Considerations

1. In general, the actions of the operators and licensee management demonstrated that they understood their plant and the procedures and philosophy for coping safely with steam generator tube rupture events....

Response: This statement is correct.

2. Given the level of training and knowledge demonstrated by the involved operators, and their philosophy that no procedure should be followed blindly, the combination of procedures utilized during the event did assist operators in making decisions that were technically adequate to safely cope with the event....

Response: We agree that the operators performed well.

3. Operators quickly determined that the plant had experienced a major steam generator tube rupture (SGTR) event based on the reactor coolant system response to the break flow and the coincidence of the air-ejector exhaust radiation monitor alarm....

Response: Confirmation of the faulted generator by means of a radiation survey is discussed in the procedure. Other statements in this finding are accurate.

4. Operators tripped the reactor coolant pumps as required by the steam generator tube rupture procedure, when reactor coolant system pressure had decreased to about 1715 psig, following the initiation of safety injection....

Response: This finding is correct. An environmentally qualified wide range pressure transmitter (P-420A) will be installed prior to plant startup and will be used for establishing the reactor coolant pump trip criteria. The trip point will be set at the appropriate (low) value. (See Reference 3.)

5. The licensee's steam generator tube rupture procedure, when followed literally, will usually result in the formation of steam in the reactor vessel upper head area for conditions of large break flows....

Response: This statement is correct. We note that analysis performed by Westinghouse, in support of the guidelines upon which the Ginna procedures are based, predicts the formation of an upper head void. The Ginna procedure has been revised to explicitly address head voiding. (See Sections 4.2.2. 4.2.7, and 8.1 of Reference 1 and Reference 3.)

6. The licensee's steam generator tube rupture procedure did not include a safety injection termination criterion which addressed the assurance of adequate core subcooling; however, the operators constantly monitored the saturation monitors and further verified core subcooling by use of the computer and by manual calculations.

Response: This statement is correct. Subcooling was, as noted, continually monitored. In addition, subcooling is now incorporated into the Ginna procedure. (See Section 8.1.1 of Reference 1 and Reference 3.)

7. Neither the licensee's steam generator tube rupture procedure nor the revision of the Westinghouse guidelines on which it was based contained instructions for coping with a failed open or leaking atmospheric PORV or Code safety valve on the faulted steam generator steam system.

Response: This statement is correct. Revision 3 of the Westinghouse Owners Group guidelines does contain these instructions.

8. During plant conditions of a pressurizer PORV stuck open, both pressurizer relief block valves shut, a steam bubble in the reactor vessel upper head region, a liquid-filled pressurizer and a faulted steam generator Code safety valve periodically relieving reactor coolant to the environment....

Response: The statement is generally correct. The steam generator code safety valve for the faulted steam generator was periodically relieving but was not, itself, faulted. The starting of the reactor coolant pump was not outside procedural guidance. (See Section 4.2 of Reference 1 for additional discussion of plant procedures.)

9. The licensee cooled the faulted steam generator by alternately feeding it with cold auxiliary feedwater and then letting it flow through the break to the reactor coolant system, which was then at a slightly lower pressure....

Response: This statement is correct. (See also Section 3.2 of Reference 1.)

10. The PORV for the B steam generator was isolated early in the event. Isolation of the PORV resulted in challenging the lowest setpoint steam generator Code safety valve with a two-phase flow condition for which it was not qualified....

Response: Isolation of the B-Steam Generator PORV is discussed in Section 4.2.5 of Reference 1. Procedure changes discussed in Section 8.1 of Reference 1. It is not clear that the safety valve failed to reseat. (See our response to Section 1.4.1, finding 11.)

11. At Ginna, the automatic switchover of the safety injection pump suction from the boric acid storage tanks to the refueling water storage tank on low level, occurs only if the safety injection signal is not reset....

Response: The desire for rapid containment isolation reset is not necessarily consistent with NRC guidance concerning containment reset in NUREG-0737. It is also not clear that containment isolation reset design resulted in any delays in operator actions since a number of other actions were being taken prior to reset being accomplished.

12. Rupture of the pressurizer relief tank rupture disc was avoidable....

Response: It is noted that only 1320 gallons of water were transferred from the pressurizer relief tank to containment sump A and that this transfer had no effect on the course of the incident. The letdown orifice valves have been modified so they will close upon containment isolation. (See Section 8.2 of Reference 1.)

13. The reactor coolant loop subcooling meter design was such that the indicated subcooling value was correct when T_{HOT} was above 500°F.. .

Response: The statement is correct. The fact that the meter indicated half the correct value when T_{HOT} was below 500°F was noted on the control board. The meter has been modified to eliminate this problem. (See Section 8.2 of Reference 1.)

14. The pressurizer PORV and block valve switches were located side by side and appeared to be identical....

Response: The statement is correct.

15. The 6-ft separation between the reactor coolant system wide-range pressure recorder and the pressurizer PORV controls required the control room operator to rely on someone else to describe the result of opening the PORV....

Response: The statement regarding control board locations is correct. However it is likely that, regardless of location of pressure indication, the operator would have employed a "bump and wait" type of operation in order to become familiar with system pressure response. Thus, it is likely that location of the pressure indication did not affect the course of the incident. It should be noted that the new wide range loop pressure transmitter

(P-420A) output is displayed on a recorder which is close to the pressurizer PORV controls.

16. A survey of the control room consoles by members of the NRC Task Force several weeks after the event, disclosed that numerous indicator lights were burned out, precluding an expeditious determination of equipment status.

Response: The indicator light status at the time of the NRC Task Force visit was not representative of operating conditions. During normal operation, all annunciator alarms are checked every shift. This insures that bulbs are not burned out. All valve position lights which indicate normal valve position are checked each shift as part of the valve line-up check. all valve status lights are reviewed regularly during plant operation. During the valve surveillance tests, which are performed as part of the pump and valve test program, typically at monthly intervals, both position indication lights for each valve and the valve status lights are checked. Any time a burned out bulb is observed during plant operation it is immediately replaced. There were no burned out light bulbs during the incident. (At the time of the NRC Task Force visit, the plant had been at cold shutdown for some time and considerable maintenance was being performed in the plant. During this condition, the safeguards systems such as the high head safety injection system and the containment spray system are not required to be operable. Any burned out bulbs would be replaced prior to systems being required to be operable.)

17. There were several examples of inconsistent use of terminology on the control panels and between the panels and the plant procedures.

Response: The statement is correct. It is noted that the inconsistencies are minor and had no affect on the course of the incident.

1.4.3 Radiological Consequences

1. It was estimated that a total of about 90 curies of noble gases were released, mostly from the steam jet-air ejector....

Response: We calculated that the bulk of tritium was released from the safety valve lift; the potential amount from

[illegible]

[illegible]

[illegible]

[illegible]

[illegible]

[illegible]

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the air ejector release was estimated to be relatively small (less than 0.05 Ci).

We concur that the staff estimate of 90 Ci of noble gas is quite conservative for the reason cited by the Task Force, which referred to the noble gases found to be trapped in the high point of the "B" steam generator subsequent to the incident.

2. During this event, the estimated releases of iodine did not exceed those calculated for the design-basis steam generator tube rupture, since the design-basis analysis assumes an iodine-spiking factor (apparently not present during the event) and a reactor coolant iodine activity concentration equal to Technical Specification limits....

Response: The statement is correct.

3. Comparison of air sample results from the licensee's fixed air-sampling stations and portable air-sampling devices, with the limits for airborne radionuclide concentrations specified in 10 CFR 20, indicated that during the event airborne concentrations in unrestricted areas offsite were less than 25% of those permissible in unrestricted areas for continuous exposure over a full year....

Response: For perspective, all releases would have also resulted in actual doses far less than the annual dose limits of 10 CFR Part 20.105. Doses onsite and offsite would have also been below the limits of 40 CFR 190, for routine operation.

4. Results of tap water samples taken by the licensee, onsite and at the Ontario Water Works, indicated no radionuclide concentrations above minimum detection capability of the instruments and procedures used.

Response: The statement is correct.

5. Because of the wide variability in area, depth, and thickness of the snow samples collected, measured concentrations of radioactive materials in snow could not be used to determine deposition quantitatively....

Response: Additional comments and information are provided in Section 7.5.2 of Reference 1.

6. With one exception, environmental surveys did not show radiation levels above background levels beyond the plant boundary....

Response: More correctly, the measurement referred to as 1.2 mrad at the intersection of Lake and Ontario Center Roads was obtained as a count rate corresponding to a dose rate of 1.2 mR/hr, and was taken on the Ginna site property at the end of the Ginna Emergency Survey Center driveway.

7. Offsite thermoluminescent dosimeter (TLD) readings by NRC and the licensee indicated that dose levels during the event were about the same as expected from background radiation sources....

Response: It should be noted that the TLD results reported as 21.7 ± 2.7 mrem and 9.4 ± 0.19 mrem are uncorrected for background.

8. The dose to the whole body of the maximally exposed individual onsite is unlikely to have exceeded about 15 mrem....

Response: Agree that 15 mrem is a conservative upper estimate. Our estimate is provided in Section 7.7 of Reference 1.

9. The dose to the whole body of the maximally exposed individual offsite is unlikely to have exceeded about 0.5 mrem....

Response: This estimate is reasonably consistent with the values provided in Section 7.7 of Reference 1, given differences in analytical approaches.

10. For the population within a 50-mile radius of Ginna, the total body population dose commitment from all pathways was estimated to be less than 0.1 person-rem.

Response: This estimate is reasonably consistent with the population exposures provided in Section 7.7 of Reference 1, given differences in analytical approaches.

11. The estimated dose to the total body of the maximally exposed individuals onsite and offsite, from radionuclides released during the event are small fractions of the average annual dose from exposure to natural background radiation in the United States.

Response: The statement is correct.

12. The total collective dose accumulated by the 146 licensee employees involved in the event on January 25 was 644 person-mrem. The maximum whole-body exposure recorded

for an individual exposed during the event was 240 millirems.

Response: The apparent 240 millirem exposure was indicated on a self-reading pocket dosimeter worn by an individual serving as a firewatch, who spent approximately 30 minutes in a low radiation dose rate area. Evaluation of readings from the other personnel monitoring devices worn by the individual indicated that the self-reading pocket dosimeter greatly overestimated the individual's exposure due to probable bumping of the dosimeter. Typical personnel exposures due to access to plant controlled areas averaged less than 25 mrem during the event.

13. The risk from exposure to radioactive materials released from Ginna is low compared with many other types of risk (radiation-related or otherwise), and the radiation-related risks are based on conservative assumptions; therefore, the risk to <u>real</u> individuals from exposure to radioactive materials released from Ginna was insignificant.

Response: The statement is correct.

14. Comparing the risk from exposure to radioactive materials released from Ginna with the risk from the normal incidence of cancer fatalities and genetic abnormalities in the general population, the risk to the public health and safety from this exposure is insignificant.

Response: The statement is correct.

15. The risks to the work-force at Ginna are very small fractions of the estimated normal incidence of cancer fatalities and genetic abnormalities.

Response: The statement is correct.

1.4.4 Institutional Response

1. The categories in which the event was classified were consistent with those in the licensee's Emergency Plan....

Response: The statement is correct in the sense of emergency de-escalation. Criteria for recovery operations are provided in RG&E's Nuclear Emergency Offsite Response Procedure. Since the incident, a Radiation Emergency Procedure has been written which addresses de-escalation of the emergency classification. This procedure is SC-110, "Ginna Station Event Evaluation for Reducing the Classification."

2. The licensee had no alternative evacuation site available for plant personnel during this event, nor does NRC require one....

Response: The statement is correct. One of our primary concerns is prompt and unconfused personnel accountability which was the rationale for the initial evacuation of personnel to the Training Center according to established procedures. Contamination of personnel was minor.

3. State and county officials jointly decided not to use the prompt notification system during this event.

Response: This statement is correct.

4. Although the licensee provided adequate notification that a steam bubble had formed in the reactor vessel head, this information was not relayed promptly by State personnel to their supervisors.

Response: The statement regarding licensee notification is correct.

5. The NRC Senior Resident Inspector functioned effectively throughout the event, particularly during the period he was the sole NRC presence at the site.

Response: We concur with this statement.

6. The results of assessments of the event by Region I Base and NRC Headquarters teams were not directly coordinated and the communications between the Resident and Region I Base teams were not always tied into the NRC Operations Center.

Response: We have no comment on this statement.

7. The Health Physics Network, Federal Telecommunications System, and commercial telecommunications systems functioned adequately, whereas the Emergency Notification System (ENS) link was only marginally acceptable....

Response: We concur with this statement.

8. The staff of the NRC Headquarters Operations Center failed to make some notifications in a timely manner, as was required.

Response: We have no comments on this statement.

1.4.5 Post-Event Activities

1. The licensee's procedure for analyses of the hydrogen and total gaseous activity of the noncondensable gases trapped in the top of the faulted steam generator was in error....

Response: During the recovery from the incident in January, several samples of entrained gas in the B Steam Generator and Reactor Head were taken. The procedure used at that time was PC-4, "Hydrogen Concentration and Radiogas Activity in Primary Coolant Sampling and Analysis". This procedure was adapted and errors were made in the expansion factor and pressure correction at the time of the initial analysis. This error was identified by RG&E personnel.

To preclude this reoccurrence, changes were made to PC-23.2, "Containment Atmosphere Sampling and Analysis During Containment Isolation". Although this procedure is designed for the containment vessel, it may be utilized for any vessel containing gasses at any pressure.

This procedure includes expansion factors and pressure corrections as well as the analytical procedures for determining hydrogen and other gas concentrations.

2. The licensee's procedure for flushing the faulted steam generator to remove dilute boric acid caused a minor deboration of the reactor coolant system....

Response: This statement is correct. (See LER 82-006 submitted on February 17, 1982.)

3. A number of foreign objects were found and removed from the secondary side of the faulted steam generator ...

Response: This subject is discussed in Section 3.5 of Reference 4.

4. The ruptured tube had a fish mouth opening 4.1 in. long and 0.7 in. wide at its maximum point, centered approximately 5 in. above the tube sheet, with the opening oriented along its tube column toward the steam generator shell....

Response: This is discussed in Section 3.3 and in Appendix A of Reference 4. We note that the wall thickness at the rupture was on the order of 5 to 8 mils, or 10% to 16% of the nominal wall thickness.

References

Ref. 1 Incident Evaluation Report, submitted by letter dated April 13, 1982 frm J. E. Maier, RG&E to Dennis M. Crutchfield, NRC.

Ref. 2 Letter dated April 26, 1982 from J. E. Maier, RG&E, to Dennis M. Crutchfield, NRC.

Ref. 3 Letter dated May 4, 1982 from J. E. Maier, RG&E, to Dennis M. Crutchfield, NRC.

Ref. 4 Steam Generator Evaluation, submitted by letter dated April 26, 1982 from J. E. Maier, RG&E, to Dennis M. Crutchfield, NRC.

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