UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

OFFICE OF NUCLEAR REACTOR REGULATION
HAROLD R. DENTON, DIRECTOR

In the Matter of	)	
	)	
ROCHESTER GAS AND ELECTRIC CORPORATION	)	Docket No. 50-244 (10 CFR 2.206)
	)	
R. E. Ginna Nuclear Power Plant	)	

DIRECTOR'S DECISION UNDER 10 CFR SECTION 2.206

By a petition dated March 11, 1982 Ms. Ruth N. Caplan, Chairman, Sierra Club National Committee, requested the Nuclear Regulatory Commission's Office of Nuclear Reactor Regulation require Rochester Gas and Electric Corporation (the licensee) to show cause why the operating license for the Ginna plant should not be suspended, or in alternative, why permission to restart the reactor should not be withheld, until such time as essential actions have been taken by the licensee and the Commission to assure the protection of public health and safety. This request has been considered under 10 CFR Section 2.206 of the Commission's regulations.

The petitioner requests that the Director of Regulation initiate a review of matters pertaining to the ability of the licensee to safely operate the Ginna plant so as to protect public health and safety in light of the January 25, 1982, steam generator tube rupture at the Ginna plant. The petitioner further requested that this review be incorporated into the review which was in progress by the staff at that time and that

DESIGNATED ORIGINAL

Certified By

it should include, but need not be limited to several specific areas discussed in the petition. Pending completion of this review, the petitioner requested that the operating license for Ginna be suspended, or in the alternative, restart of the reactor should not be permitted.

I have reviewed the information submitted by Ms. Caplan and other relevant information bearing on the issues addressed in the petition.

I

The petitioner's request that the ongoing staff safety review include and consider the specific areas detailed in the petition is granted. Many of the specific issues are addressed in the staff's Safety Evaluation Report (NUREG-0916). A reference to NUREG-0916 or a discussion of each item follows.

Petitioner's Assertion and Request

1. The cause of the tube break initiating the January 25, 1982, accident should be throughly explained and corrective action taken to prevent such breaks in the future. The mechanical damage arising from loose pieces of metal should be studied in the context of the generic corrosion problems at Ginna. Specifically, corrosion arising from AVT (all volatile treatment) control of secondary water chemistry should be addressed in relation to denting of tubes, stress corrosion, and intergranular attack. This should include corrosion in the feed-water system and corrosive impurities introduced by condenser leaks.

Response:

These issues are discussed in Sections 5.2, 5.3 and 5.4 of NUREG-0916.

Petitioner's Assertion and Request

2. The adequacy of the steam generator tube testing program should be evaluated and a determination made regarding the following issues:

   a. Is the routine multi-frequency eddy current testing method being employed at Ginna the best available given current state-of-the-art? If not, what justification is there for not employing the best available technology, in light of chronic tube degradation problems at Ginna and at other PWR's and the existence of techniques such as fiber optic examination?

   b. Is the frequency of required testing of tubes sufficient to prevent future tube rupture or other serious break?

   c. Does the current testing program, which only tests a sample of tubes and which does not test their full length, provide sufficient information to prevent tube failure?

3. The technical specifications defining the extent of allowable tube degradation for steam generator tube rejections should be reviewed in light of the Ginna accident to determine whether they are sufficiently stringent to prevent a tube break.

4. The increased risk of steam generator tube breaks/leaks, if RG&E operates the reactor without having proceeded with the preventative sleeving program originally scheduled for the Spring, 1982, refueling outage, should be assessed and a determination made as to whether the original schedule should be adhered to.

Response:

These issues are addressed in Section 5.2.4 of NUREG-0916.

Petitioner's Assertion and Request

5. The safety implications of current and proposed plugging and sleeving of steam generator tubes and of further repairs such as insertion of stabilizing cables should be examined in order to assess additional stress, such as from changes in fluid dynamics, which may be induced in tubes remaining in use.

Response:

These issues are addressed in Section 5.5.7 of NUREG-0916.

Petitioner's Assertion and Request

6. An evaluation should be completed to determine the safety implications of operator action currently required to re-establish the instrument air system and to open the PORV manually.

Response:

This issue is addressed in Section 4.2.3 of NUREG-0916.

Petitioner's Assertion and Request

7. The safety implications of the failure of the PORV to close should be assessed in light of the problems which developed during the Ginna accident, particularly with regard to the creation of a steam bubble in the reactor vessel as a result of depressurization. The potential for uncovering the core, due to a steam bubble in the reactor vessel or elsewhere in the primary system should be addressed. A determination should be made as to whether safety functions performed by the PORV required that it be designated as safety grade and be required to meet all NRC regulations applicable to such safety grade designation, in order to assure safe operation of the reactor.

Response:

Current Commission policy does not require that the PORV and its solenoid operated air valves be designated to be safety grade equipment. The staff has a generic study underway to determine whether PORVs should be required to be safety grade. The PORVs at Ginna will be considered along with all others at the completion of that evaluation. Additional information regarding the installation and operation of the PORV and void formation are contained in Sections 3.3, 4 and 6.1 of NUREG-0916.

Petitioner's Assertion and Request

8. A determination should be made, given the demonstrated unreliability of the PORV, as to whether a reliable method exists for removing decay heat by means of the secondary system, without providing, at the very minimum, one pathway for removing decay heat which consists of safety grade equipment. Such determination should also include an assessment of the reliability of essential auxiliary support systems such as instrument air, and should consider the consequences of loss of off-site power to detemine whether General Design Criteria #17 of 10 CFR Part 50 Appendix A is met.

Response:

The ability of the installed systems at the Ginna plant to provide for a reliable method for removal of decay heat was assessed by the NRC staff. The results of that review are provided in a safety evaluation issued on September 29, 1981, as part of the Systematic Evaluation Program (SEP) review of Topic VII-3, "Systems Required for Safe Shutdown." A copy of that evaluation is attached.

Petitioner's Assertion and Request

9. A determination should be made as to whether the emergency operator procedures set forth in "Westinghouse Emergency Operator Guidelines for Steam Generator Tube Rupture Events" are adequate to protect the public health and safety. Operator delay, or apparent hesitancy, in terminating the HPI (high pressure injection) is of particular concern in relation to the risk of over-pressurization of the reactor pressure vessel as reported in the Speis memorandum (see infra #11) and to the increased reliance on proper functioning of steam generator safety valves. Further, the Ginna emergency procedures should be conformed to the Westinghouse guidelines.

Response:

Since the TMI-2 accident, the staff has been actively reviewing the Westinghouse Emergency Operator Guidelines for steam generator tube ruptures. While the original guidelines from which the Ginna procedures were developed did not specifically address the possibility of a stuck open PORV, the most recent guidelines issued by Westinghouse developed in response to TMI Action Plan item I.C.1, include the consideration of multiple failures, such as PORVs failing open. They also address the possible formation of voids in the reactor vessel. While we have not yet completed our review of these guidelines, we believe they are sufficiently complete that preliminary implementation can begin. We intent to advise the W Owners of this shortly.

With respect to the adequacy of the plant specific procedures in place at the Ginna plant today, the staff evaluation of these procedures is provided in Section 4.2 of NUREG-0916.

Petitioners Assertion and Request

10. The conditions under which the reactor vessel can become over-pressurized in the course of operator action to control an accdient should be clearly specified and a determination made as to whether an automatic response system would decrease the chance of over-pressurization problems from developing and whether the installation of such a system at Ginna is an action that "will provide substantial, additional protection which is required for the public health and safety...." as provided in 10 CFR 50.109.

Response:

This issue is addressed in Section 4.2.9 of NUREG-0916.

Petitioner's Assertion and Request

11. The concerns raised in the Speis memorandum (Themis Speis to Roger Mattson, "Preliminary Evaluation of Operator Action for Ginna SG Tube Rupture Event" dated January 28, 1982, see infra Attachment E) regarding problems and potential problems in cooling the reactor following the tube break should be addressed; a determination made as to their safety significance; and necessary corrective action taken. These include the following problems:

    a. the apparent stratification in the B steam generator and its effect on slowing depressurization of the faulted steam generator;

    b. the consequence of an additional coolant system failure, including a leak in the A steam generator or "a secondary side safety/relief valve" sticking open;

    c. the necessity to remove decay heat from the A steam generator by steaming to the atmosphere due to improper functioning of the condensor;

d. the problems associated with the use of the PORV for coolant discharge during "feed and bleed" cooling.

Response:

The issues raised by items a, b, and c are addressed in Section 4.2.8, 4.2.11, 4.2.12 and 8.1 of NUREG-0916.

With regard to item d, had a leak developed in the second ("A") steam generator at Ginna, the need to institute the "feed and bleed" process to assure continued core cooling would have depended upon the leak size and total leak rate of primary coolant out of the primary system.

The staff has been evaluating the capability of operating plants to "feed and bleed" on a generic basis, although no detailed thermal-hydraulic analyses of feed and bleed have been performed for Ginna.

Limited detailed thermal hydraulic analyses have been performed by the industry however, which have shown that feed and bleed is calculated to effectively remove decay heat if sufficient HPI injection and PORV/safety valve relieving capacity is available. These analyses include (1) typical CE (e.g., Calvert Cliffs) plant; (2) B&W 177 FA plant; and (3) Sequoyah Plant (W design).

Recently, the staff evaluated the capability of all operating plants to "feed and bleed" based on each plant's HPI pump capacity and PORV/safety valve relieving capacity. Our evaluation of Ginna concluded that the Ginna plant design has sufficient PORV relieving capacity to depressurize the primary system to below the shutoff head (1475 psi) of the HPI pumps and sufficient HPI pumping capacity to remove decay heat. However, the staff points out that "feed and bleed" cooling is not a design requirement for the plant.

At Ginna, there are procedures in place which instruct the operator on how to reset the safety injection signal in order to enable reestablishing the air supply necessary for PORV operability. The procedure was, in fact, used in reestablishing instrument air which allowed the initial operation of the PORV at Ginna during the tube rupture event.

Additionally, there is a backup nitrogen system which is manually controlled from the control room which can be used to actuate the PORVs in the absence of normal instrument air.

Petitioner's Assertion and Request

12. A determination should be made as to the extent to which failure to implement the TMI Action Plan requirement for instrumentation to allow direct measurement of the water level in the reactor vessel contributed to operator problems in determining proper timing for operating the ECCS pumps and in determining the size of the steam bubble.

Response:

There are several types of water level indication systems being considered by industry and the NRC staff with respect to assisting the operator in making determinations of inadequate core cooling. Some of these systems include level indication in the reactor vessel head region. Had such a measuring device been installed, it likely would have been an aid to the operator. The operators, however, did use the available instrumentation (pressurizer level, reactor coolant system pressure, and vessel upper head thermocouples) in making determinations of the existence of the steam bubble in the reactor vessel head. Furthermore, the core exit thermocouple readings in conjunction with the reactor coolant pressure confirmed that the steam bubble was confined to the reactor vessel head area and the operator's took actions accordingly.

Petitioner's Assertion and Request

13. A full investigation should be made to determine the state of embrittlement of the Ginna reactor pressure vessel to determine the likelihood that over-pressurization will lead to vessel rupture as a consequence of pressurized thermal shock.

Response:

This issue is addressed in Section 3.5 of NUREG-0916.

Petitioner's Assertion and Request

14. The NRC should determine whether the reactor can operate safely without replacement of the steam generator and associated parts of the nuclear steam supply system and whether the newest Westinghouse steam generator design will ameliorate the problems, given the recent problems which have developed with this design at McGuire and at European reactors.

Response:

The issue of steam generator integrity and the results of our evaluation are addressed in Section 5 of NUREG-0916. Based on our conclusion, we see no need at this time to require replacement of the steam generator. We therefore consider no response necessary to the second part of this request.

Petitioner's Assertion and Request

15. The total projected worker exposure should be calculated in advance of NRC approval of RG&E's repairs and a specific plan developed to keep worker exposure as low as reasonably achievable (ALARA). This

should include a determination as to whether time should be allowed for radioactive decay, particularly of Cobalt 58, in the steam generator prior to repairs, in order to prevent unnecessary worker exposure and still allow all necessary repairs to be made.

Response:

In the course of discussions between RG&E and the staff immediately after the event, the licensee estimated that the radiation exposure incurred in the steam generator inspection and repair would be approximately 300 to 350 person-rem. The licensee described his plans to keep exposures as low as reasonably achievable, which included the use of remotely operated tools, extensive pre-planning of evolutions, and practice on special mockups. Members of the regional staff closely monitored the repair efforts to ensure that exposure was kept to a minimum, and as a result, the total exposure incurred in the repair effort was 350 person-rem. The total exposure for the entire outage is expected to be approximately 600 person-rem, which is only slightly higher than the exposure which would be typical for an outage of this magnitude without the additional steam generator repair effort. This exposure is within the expected range for PWR outages.

Petitioner's Assertion and Request

16. An overall safety assessment should be performed before the reactor is allowed to re-start in order that the combined risk of potential failure modes can be determined, in relation to the protection of public health and safety. At a minimum such an assessment should address the following:

a. the degradation of the Ginna steam generators, including the plugging, sleeving and other repairs required to date and planned;

b. the on-going contribution to tube degredation of corrosion arising from AVT control, from condenser leakage, and from the feedwater system (as apposed to the suspected damage from loose pieces of metal in the B steam generator);

c. The lack of a safety grade pathway in the secondary system to remove decay heat;

d. the chance that operator error will lead to over- or under-pressurization of the reactor vessel;

e. the state of reactor vessel embrittlement.

Response:

This request is a summary of several previous items. NUREG-0916 provides a detailed evaluation of item a, b, d and e, along with an overall, integrated assessment of their safety significance. Specifically, Sections 1, 2 and 9 address the contribution by these items to the overall risk to the health and safety of the public posed by the Ginna facility. The SEP evaluation addresses item c. The staff has reviewed these individual assessments and concludes that the return to operation of the R. E. Ginna Nuclear Power Plant is acceptable.

II

The petitioner's request that the staff issue a formal order to suspend the Ginna operating license pending evaluation of safety issues bearing on restart of a formal order was unnecessary to ensure that the licensee did not resume operation until the staff performed its safety evaluation and necessary steps

were taken to ensure adequate protection of public health and safety. As a result of a meeting on February 10, 1982 between the staff and representatives of the licensee, and other subsequent discussion, Rochester Gas and Electric Corporation agreed to to provide a complete evaluation of the event and a basis for restart of the Ginna plant. The licensee further agreed that this information would be submitted for review and approval by the staff prior to restart. This commitment was confirmed in a letter (copy attached) to the licensee from the Director of the Division of Licensing on February 24, 1982.

In light of this commitment by the licensee to delay restart until receipt of approval by the staff, the issuance of a show cause order or the suspension of the license was unnecessary. The Ginna plant has remained shut down pending approval by the staff for restart, and no formal action has been necessary to enforce the licensee's commitment.

A copy of this decision will be filed with the Secretary for the Commission's review in accordance with 10 CFR 2.206(c). As provided in this regulation, the decision will become the final action of the Commission twenty-five (25) days after issuance, unless the Commission, on its own motion, institutes review of the decision within that time.

Harold R. Denton

Harold R. Denton, Director
Office of Nuclear Reactor Regulation

Dated at Bethesda, Maryland,
this 22nd day of May, 1982.

Attachments:

1. NRC letter dtd. September 29, 1981 from D. Crutchfield to J. Maier, RG&E enclosing staff's evaluation related to Safe Shutdown Systems.
2. NRC letter dtd. February 24, 1982 from D. Crutchfield to J. Maier, RG&E relating to Ginna Steam Generator event evaluation and basis for restart.