June 18, 2001

LICENSEE: Rochester Gas and Electric Corporation

FACILITY: R. E. Ginna Nuclear Power Plant

SUBJECT: SUMMARY OF MEETING ON MAY 29, 2001, WITH ROCHESTER GAS AND ELECTRIC CORPORATION REGARDING A PROPOSED INCREASE OF CONTAINMENT PRESSURE DURING A STEAMLINE BREAK AT THE GINNA NUCLEAR POWER PLANT (TAC NO. MB1948)

On May 29, 2001 representatives of the Rochester Gas and Electric Corporation (RG&E or the licensee) met with the members of the NRC staff in Rockville, Maryland. RG&E had requested the meeting in order to brief the NRC staff on an option to correct a non-conservatism in its steamline break methodology. RG&E's evaluation of a steamline break with a limiting single active failure of the feedwater regulation valve to close indicated that containment pressure would increase above the Ginna 60 psig design pressure. Current corrective actions include cycle-specific use of reactivity feedback coefficients, as well as a limitation on the use of the full range of currently analyzed Reactivity Coolant System average temperature (TAVE).

As a result of the current corrective action, the analyzed post-steamline break pressure is below 60 psig, but RG&E has determined that it would be a significant benefit for Ginna to regain the operational flexibility relative to reactive feedback coefficients and TAVE. This could be accomplished by increasing the allowable pressure in the containment following a steamline break from 60 psig to 69 psig. The 10 CFR 50, Appendix J, Option B and the Ginna Technical Specifications definition of design pressure would, however, not change and would remain at 60 psig. The RG&E staff is confident that the increase pressure to 69 psig is acceptable considering the testing of the containment in 1969 and 1996 at 69 psig and 72 psig, respectively. The response of the containment structure at 72 psig was less than the response at 69 psig. This difference was attributed to the fact that containment concrete structural strength increases with age. The bases for the 60 psig was determined for the loss-of-coolant accident (LOCA) and not for the steamline break accident. However, since the design pressure at 60 psig has consistently been identified, the licensee believed it could not raise the pressure just for the steamline break accident and not for the LOCA without NRC acceptance and further tests. The NRC staff recognized the problem and could not immediately identify a precedence that could apply. A list of attendees is provided as Enclosure 1 and a copy of the handouts provided by RG&E is provided as Enclosure 2.

The NRC staff committed to review past history of other plants, such as St. Lucie, for examples that may apply and provide this information back to the licensee. The licensee will continue to review their options.

/RA/

Guy S. Vissing, Senior Project Manager, Section 1
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-244

Enclosures: As stated

cc w/encls: See next page

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ATTENDANCE LIST

MEETING WITH ROCHESTER GAS AND ELECTRIC CORPORATION (RG&E)

CONCERNING

A PROPOSED PLAN TO INCREASE THE GINNA ALLOWABLE CONTAINMENT

PRESSURE DURING A STEAMLINE BREAK ACCIDENT

MAY 29, 2001

Name	Organization
Guy S. Vissing	NRC/NRR/PDI-1
Peter Bamford	RG&E
Mike Ruby	RG&E
Richard Lobel	NRC/NRR/DSSA
Hans Ashar	NRC/NRR/DE
Len Sucheski	RG&E
George Wrobel	RG&E
Kamal Manoly	NRC/NRR/DE
Yong Kim	NRC/NRR/EMEB

Enclosure 1