

October 6, 2003

Dr. Robert C. Mecredy
Vice President, Nuclear Operations
Rochester Gas and Electric Corporation
89 East Avenue
Rochester, NY 14649

SUBJECT: CLOSEOUT OF RESPONSES TO GENERIC LETTER 96-06, R. E. GINNA
NUCLEAR POWER PLANT (TAC NO. M96814)

Dear Dr. Mecredy:

On September 30, 1996, the U.S. Nuclear Regulatory Commission (NRC) issued Generic Letter (GL) 96-06, "Assurance of Equipment Operability and Containment Integrity During Design-Basis Accident Conditions." In the GL, the NRC staff requested licensees to determine: (1) if containment air cooler cooling water systems are susceptible to either water-hammer or two-phase flow conditions during postulated accident conditions, and (2) if piping systems that penetrate the containment are susceptible to thermal expansion of fluid so that over-pressurization of piping could occur.

In letters dated January 30, 1997, July 21, 1998, and November 26, 2002, Rochester Gas and Electric Corporation (RG&E), provided its response to GL 96-06 for R. E. Ginna Nuclear Power Plant (Ginna).

Water-hammer or Two-Phase Flow

Cooling water systems serving the containment air coolers may be exposed to the hydrodynamic effects of water-hammer during either a loss-of-coolant accident (LOCA) or a main steamline break. In addition, cooling water systems serving the containment air coolers may experience two-phase flow conditions during design-basis accident scenarios whereas the heat removal assumptions were based on single-phase flow conditions. Therefore, cooling water systems may need corrective actions to satisfy system design and operability requirements. In GL 96-06, the NRC staff requested licensees to evaluate cooling water systems that serve containment air coolers to assure that they are not vulnerable to water-hammer and two-phase flow conditions.

Subsequent to issuance of GL 96-06, the Electric Power Research Institute (EPRI) developed an analytical methodology for evaluating the GL 96-06 water-hammer issue that was documented in EPRI Technical Reports 1003098 and 1006456 (previously known as EPRI Report TR-113594), and approved by the NRC in an evaluation dated April 3, 2002 (included as Appendix A to EPRI Technical Report 1003098). Section 3.3 of the NRC staff's evaluation requested that licensees who chose to use the EPRI methodology provide additional information to confirm that the methodology was applicable for their specific application and that it was being applied properly, and to justify any proposed exceptions.

RG&E provided the additional information that was requested in the NRC staff's evaluation of the EPRI methodology in a letter dated November 26, 2002. Your submittal indicated that the affected plant piping, components, and supports were acceptable and satisfied the applicable stress criteria and, therefore, Ginna was not vulnerable to the water-hammer and two phase flow concerns discussed in GL 96-06. While we are satisfied with your response and consider the water-hammer and two-phase flow elements of GL 96-06 to be closed, we have not performed a detailed review of your analyses; thus, they could be the subject of a future NRC audit or inspection activity.

Based on the above, the NRC staff concludes that your response adequately addresses the water-hammer and two-phase flow concerns identified in GL 96-06 for Ginna.

Thermally-Induced Over-Pressurization

Thermally-induced over-pressurization of isolated water-filled piping sections in containment could jeopardize the ability of accident-mitigating systems to perform their safety functions and could also lead to a breach of containment integrity via bypass leakage. Corrective actions may be needed to satisfy system operability requirements.

In your submittal of January 30, 1997, you identified six penetrations potentially vulnerable to a water solid volume that may be subjected to an increase in pressure due to heating of trapped fluid. The affected six lines were: Pressurizer Relief Tank Makeup Water Line; Reactor Coolant System Loop B Hot Leg Sample Line; Pressurizer Liquid Sample Line; Pressurizer Steam Sample Line; Fire Service Water Line; and, Demineralized Water Line. You determined that the affected lines were operable based on potential leakage through packing, bonnet gaskets, and/or valve seating surfaces. You also committed to take temporary actions to insure operability either by draining the line or opening a vent valve inside the containment. Also, in your Licensee Event Report (LER) submittal of January 22, 1997, you reported that the Containment Spray Charcoal Filter Dousing Line inside the containment may be subjected to an increase in pressure due to heating of trapped fluid. Your corrective action was to install a relief valve on this line which was completed on January 3, 1997.

As a permanent corrective action for the six penetrations susceptible to thermal over-pressurization, you committed in the January 30, 1997, submittal, to install relief valves on the Pressurizer Relief Tank Makeup Water Line, the Fire Service Water Line, and the Demineralized Water Line prior to the November 1997 refueling outage and on the Reactor Coolant System Loop B Hot Leg Sample Line, the Pressurizer Liquid Sample Line, and the Pressurizer Steam Sample Line during the November 1997 refueling outage. These modifications were completed.

The NRC staff finds that your corrective actions are reasonable and provide an acceptable resolution for the thermally-induced pressurization of piping runs penetrating the containment.

R. Mecredy

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The NRC staff, therefore, finds RG&E evaluations concerning thermal over-pressurization and two-phase water hammer for Ginna to be acceptable and that your response adequately addresses the concerns identified in GL 96-06.

Sincerely,

/RA/

Robert Clark, Project Manager, Section I
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-244

cc: See next page

R. Mecredy

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The NRC staff, therefore, finds RG&E evaluations concerning thermal over-pressurization and two-phase water hammer for Ginna to be acceptable and that your response adequately addresses the concerns identified in GL 96-06.

Sincerely,

/RA/

Robert Clark, Project Manager, Section I
Project Directorate I
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Office of Nuclear Reactor Regulation

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