March 29, 2002

LICENSEE: Rochester Gas and Electric Corporation (RG&E)

FACILITY: R. E. Ginna Nuclear Power Plant

SUBJECT: SUMMARY OF MARCH 22, 2002, CONFERENCE CALL REGARDING RG&E RESPONSE TO NRC BULLETIN 2002-01 "REACTOR PRESSURE VESSEL HEAD DEGRADATION AND REACTOR COOLANT PRESSURE BOUNDARY INTEGRITY" (TAC NO. MB4548)

On March 22, 2002, a conference call was held with members of the U.S. Nuclear Regulatory Commission (NRC) staff and representatives from Rochester Gas and Electric Corporation (the licensee), for Ginna Nuclear Power Plant. The list of participants is enclosed. The purpose of this call was to discuss the licensee's inspection plans in response to NRC Bulletin 2002-01. At the time of the call, the plant was in the middle of a refueling outage and was planning to begin inspection of the reactor vessel head the week of March 24, 2002.

The reactor vessel head insulation located within the shroud support ring (i.e., the center region of the head) is not readily removable from the vessel. The insulation is block insulation and the joints are sealed with a layer of cement. A waterproofing coating was placed over the insulation/cement. Although there are some cracks and minor exposed sections in the cement coating, the licensee indicated that the insulation and coating are largely intact. As a result of the above, the licensee does not plan to perform a 100-percent bare metal examination of the reactor vessel head during this outage, but will instead perform the inspections discussed below.

The licensee is planning to perform a visual inspection of the insulation on top of the reactor vessel head inside the shroud support ring. The inspection will look for any signs of boric acid leakage from the control rod drive mechanism (CRDM) housing which could cause boric acid accumulation on the reactor vessel head. Pictures will be taken of the insulation in areas adjacent to penetration nozzles to identify if any boric acid crystals are present and if deformation of the insulation has occurred. The licensee asserted that a potential through-wall crack in a penetration nozzle, j-groove weld or leakage from above the reactor vessel head would lead to an accumulation of boric acid and corrosion products at the head/insulation interface, in the annulus between insulation and nozzle, or above the insulation. The licensee also asserted that if there was corrosion occurring under the insulation, the corrosion products would push up on the insulation providing visual indication that an anomaly is present. If during the inspection should any of these conditions occur (i.e., boric acid crystals present or distortion of the insulation), the licensee will remove the insulation for further inspection. In addition, ultrasonic testing (UT) will be performed to verify the thickness of the reactor pressure vessel head for the center penetration nozzle from beneath the head. The inspection area will cover approximately two nozzle diameters. Any areas identified by UT examination to be significantly thinner than design wall thickness will be subject to additional examination, up to including insulation removal.

The NRC staff asked the licensee if they could cite any industry history that indicated that boric acid leakage due to a potential through-wall crack in a penetration nozzle, j-groove weld or leakage above the head would push up on the insulation. The licensee could not cite any industry history or experience in which this occurred. The NRC staff asked this question since it is not clear whether the boric acid and/or corrosion products could spread along the head and only cause slight lifting of the insulation which may not be noticeable during a visual inspection of the insulation. In response, the licensee stated that if this were to occur, the boric acid/corrosion products would flow by gravity towards the outer perimeter of the reactor vessel head where it would be detected outside the shroud support ring because the insulation in this region of the head is removed during refueling.

The NRC staff also asked the licensee if there had ever been any boric acid leakage that could affect reactor vessel head integrity. If leaks did occur, were the spills cleaned up upon discovery and the area inspected? Could the boric acid spill onto the insulation? If so, is the insulation permeable or are there gaps/rips in the installation? If so, what actions are planned to assess the condition of the head (e.g, removal of insulation, boroscope, UT thickness measurements, etc.)

The licensee identified three historical events associated with CRDM leakage. In 1971, a pin hole developed at the top of one CRDM housing approximately 15 feet above the head. Visual inspection at the time of discovery indicated that the leakage was localized. The area of exposed stainless steel was cleaned and the licensee indicated that no boric acid reached the reactor vessel head. In 1985, conoseals on three instrument ports leaked during a refueling outage due to improper installation. The instrument ports are located on the outer perimeter of the reactor vessel head just inside the shroud support ring. The exposed areas were cleaned and wiped down. However, it was not clear to the staff how the licensee could fully clean-up a boric acid spill inside the shroud support ring if there are gaps between the insulation and instrument ports and cracks in the insulation. The licensee stated that the joints in the block insulation are sealed with FiberFrax cement and that the insulation is coated with a silicone resin (waterproofing). Although some cracks and minor exposed sections have developed in the cement coating, the insulation and coating are largely intact; therefore, the potential exposure of the base metal of the reactor vessel head to boric acid is minimal. In 1991, seepage occurred on one of the instrument ports due to a defective seal. Inspection at the time of discovery indicated that there were no boric acid deposits on the reactor vessel head.

The licensee indicated that it did not plan to assess the condition of the reactor vessel head in the vicinity of the spills unless there was some visual indication as described above to indicate that a problem may exist. However, the licensee is planning to perform UT examination to verify the thickness of the reactor pressure vessel head for the center penetration nozzle from beneath the head as mentioned previously. In addition, the licensee also plans to perform UT examination of the external surface of the reactor pressure vessel head outside the shroud support ring on the downhill side of the four instrument ports.

With respect to inspection of the vessel head penetration nozzles for cracking, the licensee indicated they had performed eddy current inspection of the nozzles in 1999 using techniques similar to those used at Millstone 2 in 1997 (a plant with a similar susceptibility to nozzle cracking as Ginna). The licensee stated that, based on the results of their inspection and the slightly cooler reactor vessel head temperature at Ginna (as compared to Millstone 2), they do not expect to find any through-wall flaws during this outage. As a result, the licensee is not planning to perform any inspections of the penetration nozzles unless visual inspection of the insulation on top of the reactor pressure vessel head indicated possible leakage from the nozzles.

/RA/

Robert L. Clark, Project Manager, Section 1
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-244

Enclosure: As stated

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R.E. Ginna Nuclear Power Plant

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March 22, 2002

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Enclosure