December 26, 2001

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

FACILITY: R. E. Ginna Nuclear Power Plant (Ginna)

SUBJECT: SUMMARY OF DECEMBER 12, 2001, MEETING ON THE STATUS OF RG&E'S TECHNICAL EVALUATION RE: GINNA STATION CONTROL ROD DRIVE MECHANISM (CRDM) NOZZLE CRACKING (TAC NO. MB2632)

On December 12, 2001, members of the U.S. Nuclear Regulatory Commission (NRC) staff participated in a public meeting held at the NRC Headquarters in Rockville, Maryland, with representatives from RG&E and a member of the public. The meeting agenda is enclosed as Enclosure 1. The list of meeting participants is included as Enclosure 2. Enclosure 3 is a copy of the slides or handouts presented at the meeting by the participants.

The purpose of this meeting was to give RG&E the opportunity to present their technical evaluation regarding the potential for CRDM nozzle cracking to occur at Ginna and, thereby, justify why no inspection of the reactor vessel head penetration nozzles would be necessary during the next scheduled refueling outage (RFO).

Mr. William Bateman, the Branch Chief of the Materials and Chemical Engineering Branch, opened the meeting by restating the staff position presented in Bulletin 2001-01 for moderate susceptibility plants that states, at a minimum, 100 percent effective visual inspection of the reactor vessel head during the next scheduled RFO should be implemented by the licensee.

The presentations and discussions closely followed the information presented in Enclosure 3 which included: (1) general plant overview, (2) plant background information, (3) eddy current testing, (4) reactor vessel head temperature, (5) crack growth rate analysis, (6) probabilistic safety assessment (PSA), (7) proposed reactor vessel head replacement strategy, and (8) summary.

The first presentation was given by Mr. Tom Marlow of RG&E who presented an overview of what RG&E was prepared to discuss during the meeting. The second presentation was given by Mr. Brian Flynn of RG&E who then presented background information relating to Ginna. Specific items presented included:

a. Ginna has the smallest number of reactor vessel head penetrations in the industry (38).
b. RG&E has been involved with many initiatives related to this issue since it first arose in 1992.
c. Ginna is considered a moderate susceptibility plant with 15 effective full-power years to the Oconee Unit 3 condition. Ginna operates at the lowest head temperature of any plant in this category.
d. Ginna insulation rests on the reactor vessel head (it is not glued) in block form with an emulsifier. It would be very difficult to remove insulation for bare metal examination and the dose rates would be high (1.2 - 1.5 R/hr). Additionally, CRDM cooling shroud duct work would require removal, which has been in place since initial plant construction.

e. Ginna is planning to replace the reactor vessel head during the 2003 RFO. Based on the technical information presented, RG&E would prefer to devote time to preparing for the head replacement versus performing a visual inspection.

A third presentation then followed by Mr. Al Butcavage and Mr. Mike Shields of RG&E concerning the 1999 Eddy-Current examination. Specific items presented included:

a. The selected vendor had extensive experience with European plants and agreed to blind Electric Power Research Institute (EPRI) demonstration tests to confirm their capabilities.
b. All 38 penetrations were examined, including the head vent and instrument ports. The inspection covered 2 inches above and below the weld area with 93% of this surface area actually inspected. The areas with the highest stresses (welds located on the downhill side of the nozzles) were 100% inspected. The largest area not inspected was 174 degrees on the uphill side of one nozzle.
c. The 1999 inspection identified no through-wall or significant inside surface cracks, and only a shallow craze indication on the inner diameter of one nozzle. Subsequent review of this data following the lessons learned from the Oconee cracks showed no change in these conclusions. A review of the probe qualification showed that a crack from the outside surface of the nozzle would be detected if it grew to within 0.042" of the inner diameter surface.

The NRC staff noted that the 1999 Eddy-Current examination only inspected the inside surface of the reactor vessel nozzle walls and not the J-groove welds or the surrounding heat affected zones where cracking has been found recently at other plants.

The fourth presentation was made by Mr. Flynn of RG&E who discussed the Ginna reactor vessel head temperature. Specific items presented included:

a. RG&E is using a conservatively high temperature for the Material Reliability Program (MRP) susceptibility ranking.
b. The head region is provided with approximately 550 gpm of bypass flow from the reactor inlet plenum.
c. Ginna has 3 thermocouples in the head region to measure temperatures. These were used for the MRP ranking and are expected to be indicative of the highest temperatures in the upper head region. This is based on work documented in WCAP-9404 (proprietary). The NRC asked RG&E to determine if this work was previously presented to the NRC and if not, see how it could be submitted. RG&E agreed to provide this information.

The fifth presentation was made by Mr. Hal Gustin (Structural Integrity Associates) who discussed the performance of Ginna specific calculations for hypothetical crack growth based on current industry experience. Specific items presented included:

a. A code allowable through-wall flaw length of 300 degrees and critical flaw length of 338 degrees was determined for Ginna utilizing Equation 5.1 of MRP-44.
b. A 3D finite element model of the head and penetration nozzles had been developed for Ginna with the subsequent generation of stress intensity factors (K). This model using elastic fracture mechanics showed the highest values to be on the downhill side of the penetration while for previously analyzed Babcock and Wilcox (B&W) plants, the uphill

side is limiting. This difference is likely due in large part to the differing J-groove weld designs for the Ginna and B&W plants. For the previous B&W plant analyses, the weld detail showed a relatively small weld volume on the downhill side, as compared to the uphill side. For Ginna, in contrast, the weld design is such that there is actually greater weld volume on the downhill side than on the uphill side. Since weld residual stress is largely driven by the weld volume (because with more weld metal shrinking upon cooling, greater stresses are produced), one would generally expect that the region with more metal volume would experience higher stresses. For Ginna, this is the downhill region, while for the B&W plants, this is the uphill side, in general.

c. Based on the above, a crack growth law was developed for the Ginna operating temperature using the results from the MRP crack growth rate panel. This crack growth law was used to evaluate two cases. The first case involved a presumed 90 degree circumferential through-wall flaw in the nozzle with the second case using a presumed 180 degree flaw. The 90 degree flaw covered most of the industry cracks to date while the 180 degree flaw bounds both the RG&E inspection data of uninspected surface area and the industry findings to date. In both cases, the time to grow to the allowable flaw size of 300 degrees starting from the 1999 inspection is well beyond the fall of 2003 RFO when RG&E plans to replace the reactor vessel head (9 years for the 90 degree flaw and 5.3 years for the 180 degree flaw).

The sixth presentation was made by Mr. Mark Flaherty of RG&E who presented a risk based examination with respect to deferring inspections during the 2002 RFO. Specific items presented included:

a. The Conditional Core Damage Probability (CCDP) for a medium break loss-of-coolant accident is 2.25E-03 and is dominated by human actions primarily related to transferring to sump recirculation. RG&E stated that it would commit to training the station operators on this scenario during the first training cycle following startup from the 2002 RFO. The NRC raised several questions concerning how the calculated CCDP differed from the Individual Plant Evaluation values. RG&E agreed to either describe the details in their submital and/or formally docket the latest version of the PSA.
b. Actual system unavailabilities for the primary systems of importance (Residual Head Removal and Safety Injection) were low.
c. While RG&E had not performed an analysis demonstrating that Ginna had sufficient clearance between the vessel and the nozzles to detect leakage, it was presented that the Ginna Technical Specifications were the most restrictive in the industry with respect to leakage detection. Also, the station operators were very aware of leakage detection and had used it in the past to detect small leaks.

The seventh presentation was made by Mr. Brian Flynn of RG&E and Mr. Rick Klarner of B&W concerning the Ginna reactor vessel head replacement strategy. Specific items presented included:

a. RG&E has awarded a contract to B&W to replace the reactor vessel head. This decision was primarily based on the technical advances being made (e.g., improved joint design and material, improved processes for minimizing residual stresses).
b. Installation of the new head is expected during the 2003 RFO. The purchase order for the new head forging has been issued.

The last presentation was made by Mr. Tom Marlow of RG&E who summarized the presentations with the conclusion that RG&E had obtained sufficient inspection data during the 1999 RFO, which coupled with the analysis performed for potential crack growth, justifies deferral of inspections during the 2002 RFO.

Mr. Bateman closed the meeting thanking the licensee for their time and effort and reiterated that no regulatory decision would be made until the staff completes their technical review of RG&E's formal submittal of a supplement to their Bulletin 2001-01 response, which is due by December 31, 2001.

***/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

Enclosures:
1. Meeting Agenda
2. List of Meeting Participants
3. RG&E's Handouts

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**GINNA STATION CRDM CRACKING TECHNICAL EVALUATION**
**AGENDA**

| Subject | Speaker | Allocated Time |
| --- | --- | --- |
| Overview/Background | B. Flynn | 15 minutes |
| 1999 Inspections | A. Butcavage<br>M. Shields | 45 minutes |
| Reactor Vessel Head Temperature | B. Flynn | 30 minutes |
| Break | | 10 minutes |
| Crack Growth Rate Analysis | H. Gustin, SI | 30 minutes |
| Probabilistic Safety Assessment | P. Riccardello, SI<br>M. Flaherty | 30 minutes |
| Reactor Vessel Head Replacement | R. Klarner, B&W | 20 minutes |
| Summary/Conclusion | T. Marlow | 5 minutes |

Enclosure 1

**GINNA STATION CRDM CRACKING TECHNICAL EVALUATION MEETING ATTENDANCE SHEET**

Wednesday, December 12, 2001
9:00 a.m. - 12:00 p.m.
Room O8-B4

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