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STAFF EVALUATION BY THE OFFICE OF NUCLEAR REGULATORY RESEARCH

RELATING TO THE GINNA INDIVIDUAL PLANT EXAMINATION

ROCHESTER GAS AND ELECTRIC CORPORATION

R. E. GINNA NUCLEAR POWER PLANT

DOCKET NO. 50-244

1.0 INTRODUCTION

On January 15, 1994, the Rochester Gas & Electric Corporation (RG&E) submitted the Ginna Nuclear Power Plant Individual Plant Evaluation (IPE) submittal in response to Generic Letter (GL) 88-20 and associated supplements. On February 23, 1996, the staff sent a request for additional information (RAI) to the licensee. Since the IPE submittal, RG&E has made significant changes to the plant, particularly during the 1996 refueling outage. These changes included steam generator replacement, conversion to improved standard technical specifications, and a change from a 12-month to an 18-month fuel cycle. To better reflect these changes, the licensee chose to revise their original IPE submittal. The revision was provided on January 15, 1997, and included the licensee's response to the staff's RAI, whose questions were developed for the original IPE submittal but were still relevant to the re-analysis. Responses to some back-end questions were also provided by the licensee separately on June 10, 1997.

A "Step 1" review of the Ginna IPE submittal was performed and involved the efforts of Brookhaven National Laboratory in the three review areas: front-end, back-end, and human reliability analysis (HRA). The Step 1 review focused on whether the licensee's method was capable of identifying vulnerabilities. Therefore, the review considered: (1) the completeness of the information, and (2) the reasonableness of the results given the Ginna, Unit 1 design, operation, and history. A more detailed review, a "Step 2" review, was not performed for this IPE submittal. A summary of staff's findings is provided below. Details of the contractor's findings are in the attached technical evaluation report appended to this staff evaluation report (SE).

In accordance with GL 88-20, Ginna proposed to resolve Unresolved Safety Issue (USI) A-45, "Shutdown Decay Heat Removal Requirements." No other specific USIs or generic safety issues (GSIs) were proposed for resolution as part of the Ginna IPE.

The submittal states that the licensee intends to maintain a "living" probabilistic risk assessment.

2.0 EVALUATION

Ginna is a 2-loop Westinghouse pressurized-water reactor (PWR) with a large dry containment. The Ginna IPE has estimated a core damage frequency (CDF) of 5E-5 per reactor-year from internally initiated events, not including the contribution from internal floods, which will be provided at a later date. The Ginna CDF compares reasonably with that of other PWR plants. According to the licensee, loss-of-coolant accidents (LOCAs) contribute 59 percent, steam generator tube rupture (SGTR) contributes 16 percent, station blackout (SBO) contributes 12 percent, transients contribute 9 percent to the CDF, interfacing systems LOCA contributes 2 percent, and anticipated transients without scram (ATWS) contributes 2 percent.

The important system/equipment contributors to the estimated CDF that appear in the top sequences are the component cooling water, the diesel generators, the residual heat removal, and the undervoltage relays. The licensee's Level 1 analysis appears to have examined the significant initiating events and dominant accident sequences.

Based on the licensee's IPE process used to search for decay heat removal (DHR) vulnerabilities, and review of Ginna plant-specific features, the staff finds the licensee's DHR evaluation consistent with the intent of the USI A-45 (DHR Reliability) resolution and is, therefore, acceptable.

The licensee performed an HRA to document and quantify potential failures in human-system interactions and to quantify human-initiated recovery of failure events. The licensee addressed both pre-initiator actions (performed during maintenance, test, and surveillance, etc.) and post-initiator actions (performed as part of the response to an accident). The analysis of pre-initiator actions considered both miscalibrations and restoration faults.

For the quantification of post-initiator human actions, cutsets with more than one human error event were reviewed to insure independence between the events. If not independent, a change to either the model or the human error probability was made to correctly model the dependency. In addition, recovery actions (that were always procedure based) were added to cutsets, as appropriate, and were quantified using the Accident Sequence Evaluation Program (ASEP) methodology.

Human errors were identified as important contributors in accident sequences leading to core damage. The licensee identified the following operator actions as important in the estimate of the CDF: operator fails to switchover to recirculation during LOCAs, operator fails to depressurize and cooldown the reactor coolant system in SGTR accidents.

The licensee evaluated and quantified the results of the severe accident progression through the use of a containment event tree and considered uncertainties in containment response through the use of sensitivity analyses. Although detailed back-end results (including radioactive releases to the environment) are not yet available, based on the licensee responses to our RAI and our knowledge of the original IPE back-end submittal, which was reviewed for this evaluation, we conclude that the approach the licensee is using in

their back-end analysis considers important severe accident phenomena, modeled in a conventional containment event tree framework, and is, therefore, acceptable.

The licensee's response to containment performance improvement program recommendations is also consistent with the intent of GL 88-20 and associated Supplement 3. For the Ginna large dry containment, these recommendations involved the licensee performing a containment walkdown to assess features important to the plant's response to severe accidents, including the potential for localized hydrogen combustion. As a result of the walkdown, the licensee found that:

1. There was a substantial potential for global mixing with good ventilation at hydrogen release points.

2. Flames that may occur at hydrogen release points would not endanger penetrations.

3. The maximum hydrogen concentration obtainable would not support deflagration-to-detonation transition, thus local detonation would be extremely unlikely.

Some important plant safety features, identified by the licensee, which may impact CDF at Ginna are:

1. The plant has feed and bleed capability which requires both PORVs.

2. There are two motor driven auxiliary feedwater (AFW) pumps and one turbine driven AFW pump. In addition, there are two motor driven standby AFW pumps, which are manually operated in case of a high energy line break which disables the preferred AFW system.

3. The reactor coolant pumps employ Westinghouse seals with dual cooling modes: charging pump injection and component water cooling of the thermal barriers. Consistent with many other PWR PRAs, both cooling modes must fail in order for the seals to fail.

4. There are important service water dependencies; for example, both diesel generators need it for cooling.

5. High pressure recirculation is provided by "piggy-backing" two RHR pumps. The switchover from injection to recirculation is manual.

6. There are two 125V dc buses, each with an associated battery and a charger. The batteries have a 6-hour capacity without load shedding.

The licensee defined three vulnerability criteria as follows:

1. The total CDF must be less than 1.0E-04 per reactor-year (otherwise a vulnerability must be present.)

2. Are there any new or unusual means by which core damage or large early release from containment can occur than as identified in other relevant probabilistic safety assessments?

3. Does any plant design, procedure, or training feature result in a contribution to core damage or large early release from containment greater than what is expected?

The licensee identified five front-end vulnerabilities based on their vulnerability criteria. The licensee is evaluating these vulnerabilities to determine if, based on; cost and risk significance, plant changes are warranted. Two of the five, items 3 and 5 below, have been resolved, as discussed below.

The five vulnerabilities under review are:

1. Relays for steam generator low-low level actuation of AFW--the relays for this signal must energize in order to actuate the AFW; however, they are currently powered by a non-safety bus which is unavailable upon loss of offsite power.

2. ISLOCA through penetration 111--a LOCA outside containment through penetration 111 fails all RHR due to the low elevation of the RHR pump pits.

3. The standby AFW system out-of-service activities--currently, both trains of this system can be taken out of service for up to 7 days; however, it is credited for providing steam generator cooling water for certain LOCAs outside containment.

4. Charging pump suction--upon loss of dc control power or instrument air, the charging pump suction line fails open to the volume control tank, which may be empty because its supply source will have been eliminated as a result of the loss of power or air.

5. Intermediate building ventilation -- the preferred AFW pumps are located in the basement of the Intermediate Building, which is ventilated via either building exhaust fans or natural circulation from a fire door opening. However, only one train of the exhaust fans are powered by the emergency diesel generators. In the July 30, 1997, telconference with the licensee, they indicated that more detailed heatup calculations have been examined since the IPE resubmittal. They show that both the turbine driven and motor driven AFW pumps will function without ventilation for 24 hours. No plant change was recommended.

In addition, the IPE identified a non-risk significant common mode dc electrical failure related to the pressurizer PORVs that was corrected during a subsequent outage.

Vulnerabilities associated with the back-end portion of the IPE submittal, if any, will be reported along with the rest of the back-end results. In this regard, the staff believes that the licensee's process for identifying

vulnerabilities is reasonable and that any back-end vulnerabilities that may exist at Ginna, Unit 1, will be discovered by the licensee.

3.0 CONCLUSION

Based on the above findings, the staff notes that: (1) the licensee's IPE is complete with regard to the information requested by GL 88-20 (and associated guidance in NUREG-1335), and (2) the IPE results are reasonable given the Ginna design, operation, and history. As a result, the staff concludes that the licensee's IPE process is capable of identifying the most likely severe accidents and severe accident vulnerabilities, and therefore, that the Ginna IPE has met the intent of GL 88-20.

It should be noted that the staff's review primarily focused on the licensee's ability to examine Ginna for severe accident vulnerabilities. Although certain aspects of the IPE were explored in more detail than others, the review is not intended to validate the accuracy of the licensee's detailed findings (or quantification estimates) that stemmed from the examination. Therefore, this SE does not constitute NRC approval or endorsement of any IPE material for purposes other than those associated with meeting the intent of GL 88-20.

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APPENDIX

CONTRACTOR TECHNICAL EVALUATION REPORT