December 21, 2000

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

SUBJECT: REVIEW OF GINNA INDIVIDUAL PLANT EXAMINATION OF EXTERNAL EVENTS (IPEEE) SUBMITTAL (TAC NO. M83624)

Dear Dr. Mecredy:

Enclosed is the NRC Staff Evaluation Report (SER) on its review of the R.E. Ginna Nuclear Power Plant (Ginna) IPEEE submittal. Included with the SER are the contractors' Technical Evaluation Reports (TERs) in the seismic and fire areas, and the staff's TER in the high winds, external flooding, and other (HFO) external events. Also enclosed is the staff's SER for the internal flooding analysis, which you submitted as part of your IPEEE.

A screening review was performed which examined the IPEEE results for their "completeness and reasonableness" considering the design and operation of the plant. On the basis of this review and further review by a senior review board (SRB), the NRC staff concluded that the aspects of seismic events, fires, and HFO events were adequately addressed. The SRB is comprised of Office of Nuclear Regulatory Research (RES) and the Office of Nuclear Reactor Regulation (NRR) staff and RES consultants (Sandia National Laboratories) with probabilistic risk assessment (PRA) expertise in external events. In addition, based on its review, the staff has concluded that the internal flooding analysis has also been adequately addressed. The staff's review findings are summarized in the enclosed SER, and the details of the contractors' and staff's findings are included as appendices to the SER.

For the seismic IPEEE analysis, Ginna is categorized as a 0.3g focused-scope plant (per NUREG-1407). Rochester Gas and Electric Corporation (RG&E) used the Electric Power Research Institute (EPRI) Seismic Margins Assessment methodology as described in EPRI NP-6041. Originally, RG&E proposed to perform a reduced-scope evaluation; however, additional seismic studies were added consistent with NUREG-1407; therefore, the staff has concluded that the seismic assessment addresses the issues that are required for a focused-scope plant. Since the seismic margins approach was used, no quantitative estimate was made for the seismic contribution to plant core damage frequency (CDF). In the fire area, you utilized EPRI's Fire Induced Vulnerability Evaluation (FIVE) methodology to perform a screening review, and then a PRA was used to estimate the fire CDF contribution. RG&E estimated the plant fire CDF contribution to be about 3E-5 per reactor-year (RY). Since Ginna was designed and constructed prior to the issuance of the 1975 Standard Review Plan (SRP), the plant was not designed according to the SRP; however, analyses were performed to determine if the plant design conforms to the 1975 SRP criteria. RG&E evaluated HFO events using the progressive screening approach described in NUREG-1407 and Generic Letter (GL) 88-20, Supplement 4, to demonstrate that the plant meets the 1975 SRP criteria. RG&E did not estimate the contribution to CDF from HFO since these events were screened out using the NUREG-1407 screening

approach. For internal flooding, RG&E first used a qualitative screening process to screen out those internal flooding events that were judged to pose insignificant risk, and then a PRA to evaluate the remaining events. RG&E estimated that the CDF contribution from internal flooding was about 3E-5/RY. In its individual plant examination (IPE) submittal, RG&E estimated that the total CDF was 5E-5/RY for all internal events except flooding.

RG&E did not explicitly state what criterion was used to define a vulnerability in the seismic area; however, RG&E did identify certain seismic-related areas of concern that were referred to as “vulnerabilities.” These areas are discussed further at the end of Section II of the enclosed SER where their resolution is also discussed. Regarding vulnerabilities in the fire area, RG&E stated that you had used the criteria given in the Nuclear Energy Institute (NEI) Severe Accident Closure Guidelines to determine the existence of fire vulnerabilities. RG&E stated that based on the NEI criteria, that there were no fire-induced vulnerabilities. However, several potential plant and procedural modifications were identified in the fire area as a result of the IPEEE. These modifications are also discussed in Section II of the enclosed SER. RG&E did not define or identify any vulnerabilities in the HFO area. However, a related improvement such as the addition of roof scuppers was being considered to reduce roof ponding during heavy rains. Also, some block wall strengthening and turbine building structural improvements had previously been made at Ginna as a result of the Systematic Evaluation Program (SEP). These improvements are discussed in the enclosed TER for HFO.

In the internal flooding area, RG&E used three criteria to define a vulnerability: (1) the internal flooding contribution to the total internal events CDF should not cause the total to exceed 1E-4/RY; (2) are there any new or unusual means associated with internal flooding by which core damage or large, early release from containment can occur other than those identified in other relevant probabilistic safety studies? and (3) does any plant design, procedures, or training feature result in a contribution to core damage or early release from containment associated with internal flooding greater than what is expected? Based on these criteria, RG&E identified two vulnerabilities with respect to internal flooding. These vulnerabilities and their resolution are discussed further in the enclosed SER for internal flooding.

In your evaluation, you have addressed generic safety issues (GSIs) GSI-131, “Potential Seismic Interaction Involving the Movable In-Core Flux Mapping System Used in Westinghouse Plants,” GSI-57, “Effects of Fire Protection System Actuation on Safety-Related Equipment,” GSI-103, “Design for Probable Maximum Precipitation (PMP),” Unresolved Safety Issue (USI) A-45, "Shutdown Decay Heat Removal Requirements," and the Sandia Fire Risk Scoping Study (FRSS) issues which were explicitly requested in Supplement 4 to GL 88-20 and its associated guidance in NUREG-1407.

On the basis of the IPEEE review, the staff concludes that your IPEEE process is capable of identifying the most likely severe accidents and severe accident vulnerabilities, and therefore, that Ginna has met the intent of Supplement 4 to GL 88-20.

In addition, RG&E’s IPEEE submittal contains some specific information that addresses the external event aspects of certain other generic safety issues (e.g., GSI-147, “Fire-Induced Alternate Shutdown/Control Room Panel Interactions,” GSI-148, “Smoke Control and Manual Fire-Fighting Effectiveness,” GSI-156, “Systematic Evaluation Program (SEP),” and GSI-172, “Multiple System Responses Program (MSRP)”). The specific information associated with each of these issues is identified and discussed in the enclosed SER. Based on the review of the

information contained in the submittal, the staff believes that the process is capable of identifying potential vulnerabilities associated with these issues. On the basis that no potential vulnerabilities associated with the external events aspects of these issues were identified at Ginna, the staff considers these issues resolved for Ginna.

If you have any questions regarding the enclosed SER, please contact Guy S. Vissing at 301-415-1441 (e-mail - gsv@nrc.gov).

Sincerely,

/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

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STAFF EVALUATION REPORT

OF

INDIVIDUAL PLANT EXAMINATION OF EXTERNAL EVENTS (IPEEE) SUBMITTAL

ON

THE R. E. GINNA NUCLEAR POWER PLANT

Enclosure

STAFF EVALUATION REPORT OF
INDIVIDUAL PLANT EXAMINATION OF EXTERNAL EVENTS (IPEEE) SUBMITTAL
ON THE R.E. GINNA NUCLEAR POWER PLANT

1.0 INTRODUCTION

On June 28, 1991, the NRC issued Generic Letter (GL) 88-20, Supplement 4 (with NUREG-1407, Procedural and Submittal Guidance) requesting all licensees to perform individual plant examinations of external events (IPEEE) to identify plant-specific vulnerabilities to severe accidents and to report the results to the Commission together with any licensee-determined improvements and corrective actions. The licensee, Rochester Gas & Electric Corporation (RG&E), provided its response to the NRC in attachments to a series of letters covering the seismic; fire; and high winds, external floods, and other (HFO) external events areas of the IPEEE. By topic, these documents were as follows: Seismic: January 31, 1997 (Reference 1); Fire: June 30, 1998 (Reference 2), and August 19, 1998 (Reference 3); HFO: September 8, 1998 (Reference 4), and December 21, 1998 (Reference 5). In addition, the licensee submitted a revised internal flooding analysis on March 1, 1999 (Reference 6) as a part of their IPEEE. This analysis was revised from the one that was included in the earlier submittal of the individual plant examination of internal events (IPE) for Ginna.

The NRC staff contracted with Brookhaven National Laboratory and Sandia National Laboratories to conduct screening reviews of the licensee's IPEEE submittal in the seismic and fire areas, respectively. The NRC staff conducted screening reviews of the licensee's submittals in the areas of HFO events and internal flooding. The staff sent a request for additional information (RAI) to the licensee on February 25, 1999 (Reference 7), and the licensee responded in a letter dated July 30, 1999 (Reference 8). Based on the results of the review of the submittals and the licensee's responses to the RAIs, the staff concluded that the aspects of seismic events, fires, HFO events, and internal flooding were adequately addressed. The review findings are summarized in the evaluation section below. Details of the staff's and contractors' findings are presented in the four attachments to this staff evaluation report (SER).

In accordance with Supplement 4 to GL 88-20, the licensee provided information to address the resolution of Unresolved Safety Issue (USI) A-45, "Shutdown Decay Heat Removal Requirements," Generic Safety Issue GSI-103, "Design for Probable Maximum Precipitation (PMP)," GSI-131, "Potential Seismic Interaction Involving Movable In-Core Flux Mapping System Used in Westinghouse Plants," GSI-57, "Effects of Fire Protection System Actuation on Safety-Related Equipment," and the Sandia Fire Risk Scoping Study (FRSS) issues. These issues were explicitly requested in Supplement 4 to GL 88-20 and its associated guidance in NUREG-1407. Staff and contractor review findings regarding these issues are included in this SER. The licensee did not propose to resolve any additional USIs or GSIs as part of the R.E. Ginna Nuclear Power Plant (Ginna) IPEEE.

An IPEEE Senior Review Board (SRB) was established and meets on a regular basis. The purposes of the SRB are (1) for the contractor to present the findings and conclusions of its review and the bases for its conclusions, and (2) for the SRB members to provide their perspectives on the contractor's findings and conclusions and to make recommendations based on their technical expertise. In this manner, the SRB provides additional assurance that (1) the scope of the review meets the objectives of the program, and (2) critical issues that have the potential to mask vulnerabilities are not overlooked.

2.0 EVALUATION

Ginna is a single-unit, Westinghouse two-loop pressurized-water reactor (PWR) with a power output of 470 MWe. The plant is located approximately 20 miles northeast of Rochester, New York on the south shore of Lake Ontario. The Ginna plant is owned by RG&E Company and began commercial operation in July 1970.

For the seismic analysis, Ginna is categorized as a 0.3g focused-scope plant (per NUREG-1407). While the licensee originally committed to perform a reduced-scope evaluation using the Electric Power Research Institute (EPRI) seismic margins assessment methodology (as described in NP-6041), additional seismic studies were added consistent with NUREG-1407 so that the staff has concluded that the licensee's seismic assessment addresses the issues that are required for a focused-scope plant. These additional studies included the listing of all safety-related structures and components and the use of the focused-scope seismic assessment approach described in NP-6041. The safety-related buildings at Ginna are all founded on rock. The plant licensing seismic design basis earthquake (DBE) was based on a Housner spectrum anchored to 0.2g; however, a 0.2g Regulatory Guide (RG) 1.60 spectrum was used for the additional seismic studies to extend the plant's seismic capacity beyond the original seismic design basis. In the fire area, the licensee utilized EPRI's Fire Induced Vulnerability Evaluation (FIVE) methodology to perform a screening review, and then a probabilistic risk assessment (PRA) was used to estimate the fire CDF contribution. Since Ginna was designed and constructed prior to the 1975 Standard Review Plan (SRP), the plant was not designed according to the SRP; however, the licensee performed analyses to determine if the plant design conforms to the 1975 SRP criteria. The licensee evaluated HFO events using the progressive screening approach described in NUREG-1407 and GL 88-20, Supplement 4, to demonstrate that the plant meets the 1975 SRP criteria. For internal flooding, the licensee first used a qualitative screening process to screen out those internal flooding events that were judged to pose insignificant risk, and then a PRA to evaluate the remaining events.

Core Damage Frequency Estimates

The licensee did not quantitatively estimate a seismic core damage frequency (CDF) contribution, since a seismic margin assessment was performed. A quantification for fire events indicated that the contribution to plant CDF from fire was about 3E-5 per reactor-year (RY). The CDF contribution from HFO events was not quantitatively estimated since these events were screened out using the NUREG-1407 screening approach. The licensee estimated that the CDF contribution from internal flooding was about 3E-5/RY. In its individual plant examination (IPE) submittal, the licensee estimated that the total CDF was about 5E-5/RY for all internal events except flooding.

Dominant Contributors

Based on the results of the licensee's seismic analysis, the components with the lowest High Confidence of Low Probability of Failure (HCLPF) values were reported in Tables 3 and 4 of Reference 1. These were the components that did not pass the screening criteria. Examples of the components that were listed in Tables 3 and 4 are turbine injection pump cooler outlet relief valves, containment recirculating filter and cooling units, batteries, auxiliary feedwater crossover motor-operated valves, and auxiliary feedwater discharge flow transmitters. HCLPF values were

not provided in the tables, so it is not possible to rank the contributors from the information provided. For fire events, the licensee reported that the main contributors to the fire-related CDF included the control, turbine, and auxiliary buildings, and the transformer yard. As indicated above, it is not possible to rank the CDF contributions of the HFO-related events since these events were not quantitatively estimated. For internal flooding, the licensee estimated that flooding in the battery rooms contributes about 77 percent to the total internal flooding CDF contribution; flooding in the screenhouse, which is assumed to fail all four service pumps, contributes about 9 percent; flooding in the relay room about 6 percent; and flooding in the auxiliary building basement contributes about 3 percent.

The licensee's IPEEE assessment appears to have examined the significant initiating events and dominant accident sequences.

Containment Performance

Plant walkdowns were performed using special procedures by the licensee (specifically including consideration of failures of mechanical and electrical penetrations) to evaluate potential seismic, fire, and flooding-induced containment failures. The licensee stated that its review of containment performance during seismic events focused specifically on issues related to structural integrity, isolation systems, bypass systems, and plant-unique containment systems. Specific items that were reviewed were all safety-related, air, motor, and solenoid operated valves, personnel and equipment hatches and the fuel transfer canal. The licensee stated that no features were identified that would lead to early containment failure. The licensee indicated that studies were also performed to assess the vulnerability of the containment to fires. Specific areas that were focused on were containment isolation and bypass, and containment heat removal. The licensee reported that based on a conservative analysis, no vulnerabilities due to fire were identified. The licensee performed an assessment of the effects of internal flooding on containment performance, particularly regarding the ability to isolate. The licensee reported that the potential for containment isolation failure due to flooding was not found to be uniquely impacted by flood events and, therefore, there were no new containment issues due to flooding conditions.

The licensee's containment performance analyses for seismic, fire, and flooding events appears to have considered important containment performance issues and are consistent with the intent of Supplement 4 to GL 88-20.

Generic Safety Issues

As a part of the IPEEE, a set of generic and unresolved safety issues (USI A-45, GSI-131, GI-103, GSI-57, and the Sandia FRSS issues) were identified in Supplement 4 to GL 88-20 and its associated guidance in NUREG-1407 as needing to be addressed in the IPEEE. These safety issues were evaluated by the NRC's contractors, and the results of these evaluations are contained in the attached TERs. For those safety issues that were not completely resolved by the contractors, the NRC staff performed additional reviews. The final resolution of these issues is provided below.

1. USI A-45, “Shutdown Decay Heat Removal Requirements”

   This issue was addressed by the licensee in Section 7.1 of Reference 1 for the seismic area, and in Section 3.10.1 of Reference 2 for fire. During the seismic margins assessment, the licensee evaluated the components that comprised the decay heat removal system and concluded that this issue presented no vulnerabilities for Ginna during seismic events. Regarding the fire aspects of USI A-45, the licensee reported that the plant’s long-term decay heat removal capability was evaluated for fire conditions and that no vulnerabilities were identified.

   The staff finds that the licensee’s evaluation of USI A-45 is consistent with the guidance provided in Section 6.3.3.1 of NUREG-1407 and, therefore, the staff considers this issue resolved.

2. GSI-131, “Potential Seismic Interaction Involving the Movable In-Core Flux Mapping System Used in Westinghouse Plants”

   The licensee addressed this issue in Section 7.2 of Reference 1 in which it is reported that the in-core flux mapping system was examined during the A-46/IPEEE Seismic Review Team during the containment walk down. The licensee stated that, based on this examination, it was concluded that the design of the mapping system, specifically including the general construction and anchorage, was sound. The staff finds that the licensee’s evaluation is consistent with the guidance provided in Section 6.3.3 of NUREG-1407 and, therefore, the staff considers this issue resolved.

3. GSI-103, “Design for Probable Maximum Precipitation”

   In Reference 5, the licensee reported that the new Probable Maximum Precipitation (PMP) criteria had been considered and that it was determined that on-site flooding and roof ponding resulting from the increased rainfall did not pose a significant threat to the plant with the exception that one building had a marginal roof design. This building was the Control Building for which the building parapet was to be modified or an additional scupper added to minimize ponding on the roof. The licensee used the National Weather Service’s revised Hydrometeorological Report No. 52 to determine that the value of PMP for the Ginna site is 16.5 inches over one-square mile in 1 hour. Overall, the staff finds that the licensee’s GSI-103 evaluation is consistent with the guidance provided in Section 6.2.2.3 of NUREG-1407 and, therefore, the staff considers this issue resolved.

4. GSI-57, “Effects of Fire Protection System Actuation on Safety-Related Equipment”

   The licensee has assessed the impact of inadvertent actuation of fire protection systems on safety systems which is one of the FRSS issues. The IPEEE submittal addresses this issue in Section 5.6.2 of Reference 1 and Section 3.9.3 of Reference 2. The licensee stated that, based on their assessment, they conclude that actuation of fire protection measures would not result in the impairment of safety-related equipment. The staff finds that the licensee’s GSI-57 evaluation is consistent with the guidance provided in EPRI’s FIVE which was accepted by the NRC staff and, therefore, the staff considers this issue resolved.

5. FRSS Issues

   In Section 3.9 of Reference 2, the licensee has explicitly addressed the FRSS issues following the EPRI guidance on these issues. These issues are: (1) seismic-fire interactions; (2) adequacy of fire barriers; (3) smoke control and manual fire fighting effectiveness; (4) equipment survival in a fire-induced environment; and (5) fire-induced alternate shutdown/control room panel interaction. The licensee states in its submittal that it has not identified any unacceptable risks or outliers at Ginna due to the FRSS issues. The staff finds that the licensee's evaluation is consistent with the guidance provided in NUREG-1407 and, therefore, the staff considers these issues resolved.

<u>Other Generic Safety Issues</u>

In addition to those safety issues discussed above that were explicitly requested in Supplement 4 to GL 88-20, four GSIs were not specifically identified as issues to be resolved under the IPEEE program; thus, they were not explicitly discussed in Supplement 4 to GL 88-20 or NUREG-1407. However, subsequent to the issuance of the GL, the NRC evaluated the scope and the specific information requested in the GL and the associated IPEEE guidance, and concluded that the plant-specific analyses being requested in the IPEEE program could also be used, through a satisfactory IPEEE submittal review, to resolve the external event aspects of these four safety issues. These GSIs were initially evaluated by the NRC's contractors, and the results of these evaluations are contained in the attached TERs. For those GSIs that were not completely resolved by the NRC's contractors, the NRC staff performed additional reviews. The final resolution of these issues is provided below.

1. GSI-147, "Fire-Induced Alternate Shutdown/Control Room Panel Interactions"

   The licensee addressed this issue in Sections 3.9.5 and 3.10.2 of Reference 2, and in their RAI response provided in Reference 7. The licensee discussed plant inspections, walkdowns, and studies that indicated that Ginna is adequately protected against potential control systems interactions under fire conditions. The licensee indicated that a review of the control and monitoring circuits of the plant was conducted to verify that safe shutdown circuits have been physically located independent of, or can be isolated from, the control room. Based on the results of the IPEEE submittal review and the review of the licensee's RAI responses, the staff considers that the licensee's process is capable of identifying potential vulnerabilities associated with this issue. On the basis that no vulnerability associated with this issue was identified in the IPEEE submittal, the staff considers this issue resolved.

2. GSI-148, "Smoke Control and Manual Fire-Fighting Effectiveness"

   The licensee's IPEEE submittal contains information addressing this issue in Section 3.9.4 of Reference 2. The licensee concluded that the Ginna fire protection systems and procedures provide adequate assurance that manual fire fighting effectiveness will not be significantly degraded from smoke and heat effects. Plant procedures establishing fire brigade training and drill requirements, evacuation plans, and inspection requirements were identified in the licensee's submittal. Based on the results of the IPEEE submittal review, the staff considers that the licensee's process is capable of

identifying potential vulnerabilities associated with this issue. On the basis that no vulnerability associated with this issue was identified in the IPEEE submittal, the staff considers this issue resolved for Ginna.

3. GSI-156, "Systematic Evaluation Program (SEP)"

   The licensee's IPEEE submittal contains information to directly address the following external-event-related SEP issues: (1) settlement of foundations and buried equipment (not required for a rock site); (2) dam integrity and site flooding (Section 5.2 of Reference 1 and Reference 5); (3) seismic design of structures, systems, and components (Ref. 1); (4) site hydrology and ability to withstand floods (Reference 5); (5) industrial hazards (Reference 4); (6) tornado missiles (Reference 3); (7) severe weather effects on structures (Reference 5); (8) design codes, criteria and load combinations (Reference 5); and (9) shutdown systems and electrical instrumentation and control features (Reference 7).

   Based on the results of the IPEEE submittal review, the staff considers that the licensee's process is reasonable and is capable of identifying potential vulnerabilities associated with this issue. On the basis that no vulnerabilities associated with this issue were identified in the licensee's IPEEE submittal, the staff considers this issue resolved for Ginna.

4. GSI-172, "Multiple System Responses Program (MSRP)"

   The licensee's IPEEE submittal contains information directly addressing the following external-event-related MSRP issues: (1) effects of fire protection system actuation on non-safety related and safety-related equipment (Section 5.6.2 of Reference 1 and Section 3.9.3 of Reference 2); (2) seismically-induced fire suppression actuation (Section 5.6.2 of Reference 1 and Section 3.9.3 of Reference 2); (3) seismically induced fires (Section 5.6.2 of Reference 1, Section 3.9.3 of Reference 2); (4) effects of hydrogen line ruptures (Section 5.6.2 of Reference 1 and Section 3.9.3 of Reference 2); (5) the IPEEE-related aspects of common cause failures related to human errors (RAI response dated July 30, 1999, for seismic events; and Sections 3.7.1 and 3.9.4 for fire scenarios); (6) non-safety-related control system/safety-related protection system dependencies (Section 5 of Reference 1 and Section 3.9.5 of Reference 2); (7) effects of flooding and/or moisture intrusion on non-safety related and safety-related equipment (Sections Section 5.6.2 of Reference 1 and Section 3.9.3 of Reference 2); (8) seismically-induced spatial/functional interactions (Section 5 of Reference 1); (9) seismically-induced flooding (Section 5.6 of Reference 1); (10) seismically-induced relay chatter (Section 5.4 of Reference 1); and (11) evaluation of earthquake magnitudes greater than the safe-shutdown earthquake (Section 7.3 of Reference 1).

   Based on the overall results of the staff's IPEEE submittal review, the staff considers that the licensee's review process is capable of identifying potential vulnerabilities associated with GSI-172. Therefore, on the basis that all potential vulnerabilities associated with these issues were reported to have been resolved by the licensee, the staff considers the IPEEE-related aspects of these issues to be resolved for Ginna.

No other specific USIs or GSIs were proposed by the licensee for resolution as part of the Ginna IPEEE.

Unique Plant Features, Potential Vulnerabilities, and Improvements

There were no unique plant features noted in the licensee's IPEEE submittal. The licensee did not explicitly define a criterion for use in identifying a seismic vulnerability; however, the licensee did state that based on their IPEEE review, a number of "vulnerabilities" were identified in the seismic area. The licensee addressed these vulnerabilities associated with seismic/fire interactions in Section 3.9.3 of its IPEEE submittal (fire) and indicated that all of these issues were resolved. In addition, the licensee stated that its seismic margin analysis indicated that there were a number of equipment components that could not be screened out using the high confidence of low probability of failure (HCLPF) value for focused review level plants of 0.3g. The licensee stated in the RAI response (Reference 7) that some of the issues associated with the unscreened components had been resolved but that others were still being evaluated.

For the fire area, the licensee stated that it compared the Ginna fire PRA results with the Nuclear Energy Institute's (NEI) Severe Accident Closure Guidelines to search for fire-induced vulnerabilities but that none were identified. However, several plant and procedural modifications were identified. A modification to install fuses on control circuits routed in the screen house associated with the 4160 VAC circuit breakers was made in 1999. Several other modifications were considered but were later dismissed after a requantification of the fire PRA indicated that the modifications were not needed. The licensee discussed these areas in its response to RAI #8 on fire (Reference 7) in which it was stated that the bases for the dismissal were the removal of unnecessary conservatism in the analyses and appropriately crediting existing procedural actions.

Regarding the HFO area, the licensee stated that it had previously made certain structural improvements to the plant to provide additional protection against high winds as a result of the Systematic Evaluation Program (SEP) review. These structural improvements resulting from the SEP are described in the R.E. Ginna final safety analysis report (FSAR) in Section 3.3. In addition, the licensee stated that it was evaluating the addition of additional roof scuppers or some other modification to the roof of the Control Building to provide additional protection against roof ponding loads.

In the internal flooding area, the licensee used three criteria to define a vulnerability: (1) the internal flooding contribution to the total internal events CDF should not cause the total to exceed 1E-4/RY; (2) are there any new or unusual means associated with internal flooding by which core damage or large, early release from containment can occur other than those identified in other relevant probabilistic safety studies? (3) does any plant design, procedures, or training feature result in a contribution to core damage or early release from containment associated with internal flooding greater than what is expected? Based on these criteria, the licensee identified two vulnerabilities with respect to internal flooding: a lack of procedural guidance for relay room internal flooding, and a lack of flood detection capability in the battery rooms A and B. The licensee reported that a procedure was modified to address the issue of relay room flooding, and plans were made to relocate the service water piping from the battery rooms to prevent flooding in those locations. In addition, the licensee reported that they had implemented an on-line risk monitor that includes a consideration of flooding-related failures. This risk monitor is used by Ginna staff to evaluate the impact of taking equipment out of service for maintenance and repair. These plant changes to address internal flooding are discussed in the attached SER on internal flooding.

3.0 CONCLUSION

On the basis of the above findings, the staff notes that: (1) the licensee's IPEEE is complete with regard to the information requested by Supplement 4 to GL 88-20 (and associated guidance in NUREG-1407), and (2) the IPEEE results are reasonable given the Ginna design, operation, and history. Therefore, the staff concludes that the licensee's IPEEE process is capable of identifying the most likely severe accidents and severe accident vulnerabilities, and therefore, that the Ginna IPEEE has met the intent of Supplement 4 to GL 88-20 and the resolution of specific generic safety issues discussed in this evaluation report.

It should be noted that the staff focused its review primarily on the licensee's ability to examine Ginna for severe accident vulnerabilities. Although certain aspects of the IPEEE were explored in more detail than others, the review was not intended to validate the accuracy of the licensee's detailed findings (or quantitative estimates) that underlie or stem from the examination. Therefore, this evaluation report does not constitute NRC approval or endorsement of any IPEEE material for purposes other than those associated with meeting the intent of Supplement 4 to GL 88-20 and the resolution of specific generic safety issues discussed in this staff evaluation report.

4.0 REFERENCES

1. IPEEE Seismic Evaluation Report, Attachment to letter from Robert C. Mecredy, Rochester Gas and Electric Corporation, to USNRC, January 31, 1997.

2. IPEEE Fire Analysis Report, Attachment to letter from Robert C. Mecredy, Rochester Gas and Electric Corporation, to USNRC, June 30, 1998.

3. IPEEE High Winds and Transportation Report, from Robert C. Mecredy, Rochester Gas and Electric Corporation, to USNRC, August 19, 1998.

4. IPEEE Supplement to High Winds and Transportation Report, Attachment to letter from Robert C. Mecredy, Rochester Gas and Electric Corporation, to USNRC, September 8, 1998.

5. IPEEE Probable Maximum Precipitation Analysis Report, Attachment to letter from Robert C. Mecredy, Rochester Gas and Electric Corporation, to USNRC, December 21, 1998.

6. Response to February 23, 1996, Request for Additional Information (RAI) (Internal Flooding Issues Only), Attachment to letter from Robert C. Mecredy, Rochester Gas and Electric Corporation, to USNRC, March 1, 1999.

7. Request for Additional Information on the R.E. Ginna Nuclear Power Plant IPEEE Submittal (TAC No. M83624), Letter from Guy S. Vissing, USNRC, to Robert C. Mecredy, Rochester Gas and Electric Corporation, February 25, 1999.

8. Response to Request for Additional Information on IPEEE, Attachment to letter from Robert C. Mecredy, Rochester Gas and Electric Corporation, to USNRC, July 30, 1999.

Attachment 1

R.E. GINNA NUCLEAR POWER PLANT

INDIVIDUAL PLANT EXAMINATION OF EXTERNAL EVENTS (IPEEE)

TECHNICAL EVALUATION REPORT

SEISMIC EVENTS

SUBMITTAL-ONLY SCREENING REVIEW
OF THE
R. E. GINNA NUCLEAR POWER PLANT
INDIVIDUAL PLANT EXAMINATION
FOR
EXTERNAL EVENTS

(Seismic Portion)

November 1998
(Updated July 2000)
(Finalized August 2000)

BROOKHAVEN NATIONAL LABORATORY

LIST OF ACRONYMS

AFW	Auxiliary Feedwater
BNL	Brookhaven National Laboratory
CCW	Component Cooling Water
DBE	Design Basis Earthquake
DHR	Decay Heat Removal
EPRI	Electric Power Research Institute
GIP	Generic Implementation Procedure
GL	Generic Letter
HCLPF	High Confidence of Low Probability of Failure
IPE	Individual Plant Examination
IPEEE	Individual Plant Examination of External Events
ISRS	In-Structure Response Spectra
LOCA	Loss-of-Coolant Accident
NRC	Nuclear Regulatory Commission
NSSS	Nuclear Steam Supply System
PWR	Pressurized Water Reactor
RAI	Request for Additional Information
RG	Regulatory Guide
RG&E	Rochester Gas and Electric Corporation
RHR	Residual Heat Removal
RLE	Review Level Earthquake
RWST	Refueling Water Storage Tank

S&A	Stevenson and Associates
SAFW	Standby Auxiliary Feedwater

SEP	Systematic Evaluation Program
SI	Safety Injection
SMA	Seismic Margins Assessment
SPLD	Success Path Logic Diagram
SQUG	Seismic Qualification Utility Group
SRT	Seismic Review Team
SSEL	Safe Shutdown Equipment List
USI	Unresolved Safety Issue

1.0 INTRODUCTION

1.1 Purpose

In response to the U.S. Nuclear Regulatory Commission (NRC) issued Supplements 4 and 5 to Generic Letter (GL) 88-20, "Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities - 10 CFR 50.54(f)," Rochester Gas & Electric Corporation (RG&E) performed an IPEEE for the R. E. Ginna Nuclear Power Plant (Ginna), and submitted the IPEEE results to the NRC on January 31, 1997 (Reference 1). Brookhaven National Laboratory (BNL), as requested by the NRC, has performed the submittal-only screening review to verify the technical adequacy of the seismic portion of the IPEEE submittal. As a result of this review, the NRC sent a Request for Additional Information (RAI) to RG&E. RG&E responded to the RAIs in Attachment A to the July 30, 1999 letter (Reference 2) to the NRC. This Screening Review presents the results and conclusions of the BNL review and evaluation of both the original submittal and the subsequent licensee's response.

BNL's methodology utilized for the review followed the guidelines provided in the document titled "Guidance for the Performance of Screening Reviews of Submittals in Response to USNRC Generic Letter 88-20, Supplement 4 " (Draft, October 24, 1996), as amended by the NRC.

1.2 Background

Ginna is a single unit, pressurized water reactor (PWR) nuclear steam supply system (NSSS) and a turbine-generator furnished by Westinghouse Electric Corporation. The containment, which houses the NSSS components, is a steel-lined, vertically prestressed and horizontally reinforced concrete structure. The power output is 470 MWe. Commercial operation commenced in July 1970. The balance of the plant was designed and constructed by Gilbert.

The safety related buildings at Ginna are all founded on rock. The plant licensing seismic design basis earthquake (DBE) was based on Housner spectrum anchored to 0.2g PGA. Ginna is binned in the 0.3g focused-scope category according to NUREG-1407.

1.3 Licensee's IPEEE Process and Licensee's Insights

Rather than performing a system analysis to define two safe shutdown success paths as outlined in Electric Power Research Institute (EPRI) NP-6041, the Ginna licensee decided to conduct a seismic IPEEE review and assessment of all safety related components in the plant. This includes buildings, active mechanical and electrical equipment, tanks and heat exchangers, piping, electrical raceways, and ducting. By letter, dated November 7, 1995 from Robert C. Mecredy (RG&E) to Allen R. Johnson (USNRC), Ginna committed to perform a reduced-scope seismic evaluation, using the EPRI methodology for Seismic Margins Assessment (SMA), as described in NP-6041. The IPEEE walkdowns were conducted together with the USI A-46 walkdowns by a team of engineers with required disciplines from RG&E and Stevenson and Associates (S&A). The results of the SMA are included in the IPEEE report that was submitted to the NRC with the letter dated January 31, 1997.

Ginna, which received its construction permit in 1966, is one of the oldest operating nuclear power plants in the United States. During its life, Ginna has undergone a number of programs addressing seismic design issues, namely:

SEP (Systematic Evaluation Program)
IE bulletin 79-02, Pipe Support Base Plate Designs Using Concrete Expansion Anchor Bolts
IE Bulletin 79-14, Seismic Analysis for As-Built Safety-Related Piping Systems
IE Bulletin 80-11, Masonry Wall Design
IE Bulletin 80-21, Anchorage of Safety Related Electrical Equipment
USI A-46, Verification of Seismic Adequacy of Equipment in Older Operating Nuclear Plants.

These programs started in the late 1970s; most were completed by the mid 1980s, except for USI A-46, which was addressed together with the IPEEE program. Under these programs, RG&E has conducted extensive reevaluations of, and made upgrades to, Ginna's structures, systems, and equipment, using a 0.2g Regulatory Guide (RG) 1.60 spectrum as seismic input. These efforts have extended Ginna's seismic capacity beyond the original seismic design basis, which is a 0.2 Housner SSE spectrum.

Originally Ginna addressed the seismic IPEEE using the EPRI methodology for a reduced scope plant, but also carried out the following:

- Screening Tables 2-3 and 2-4 of EPRI NP-6041 for the focused-scope plants were initially applied. If a component failed to satisfy the screening criteria as listed in these tables, its seismic evaluation would be based on the existing requirements from the previous programs as discussed above.

- All safety-related components were walked down instead of a specified subset of equipment expected to be required following a seismic event.

In addition, the licensee's RAI's response provided more information and discussions to address the issues that are required for a focused-scope plant. Ginna listed all safety related structures, components, and performed the seismic evaluations based on the Screening Tables 2-3 and 2-4 of EPRI NP-6041 for focused-scope plants and the 0.2g RG 1.60 spectrum. In light of both the IPEEE submittal and the RAI response, the overall process, methods, and organization of the licensee's IPEEE program appear satisfactory with respect to the requested information outlined in NUREG-1407.

2.0 REVIEW FINDINGS

2.1 IPEEE Format and Methodology Documentation

Ginna did not perform a system analysis for the safe shutdown paths. Instead, all safety-related equipment was included for the seismic evaluation. The seismic IPEEE submittal addressed plant walkdowns, structural analysis and component screens, seismic/fire and flood interactions, spatial interactions, and containment performance. Issues, such as system analysis and nonseismic failures and human actions, which were not

discussed in the submittal, were addressed in the licensee's RAIs response (Reference 2). Discussions were provided with respect to certain generic issues identified in NUREG-1407.

Although the submittal was not organized in the sample format provided in NUREG-1407, and the method and associated assumptions used to address the seismic IPEEE issues are inconsistent with the guidelines of NUREG-1407 for a focused-scope plant, the IPEEE format and methodology documentation appears satisfactory.

2.2 Seismic Review Team Selection

The same Seismic Review Team (SRT), which performed the walkdown for the USI A-46 analysis, conducted the seismic evaluations for the IPEEE program. The SRT was comprised of engineers with required disciplines from RG&E and Stevenson and Associates. Credentials of SRT members are provided in the USI A-46 submittal, which is an attachment to the IPEEE submittal. The SRT selection is in full compliance with requirements of EPRI NP-6041. The submittal did not mention whether an independent peer review of the IPEEE process was performed.

2.3 Seismic Input

The plant original licensing seismic design basis earthquake (DBE) was based on a 0.2g Housner spectrum. The plant safety related structures and components were later upgraded to a 0.2g RG 1.60 spectrum under SEP and other programs as described in Section 1.3. All safety-related structures are founded on rock.

The review level earthquake (RLE) assigned to the plant by the NRC is 0.3g and the plant is binned in the focused-scope category.

2.4 Success Path Selection and Safe Shutdown Equipment List (SSEL)

In the Ginna IPEEE submittal it is simply stated (in Section 2 System analysis) that "Rather than performing a system analysis to define the scope of the review, this assessment includes all safety related components in the plant. This includes buildings, active mechanical and electrical equipment, tanks and heat exchangers, piping, electrical raceways, and ducting." Success path logical diagrams (SPLDs) such as those described in EPRI NP-6041 are not developed and discussed in the Ginna IPEEE submittal, but are discussed in a report attached to the IPEEE submittal and the licensee's response to the RAI.

The SPLD for one success path is provided in a "Report on the Development of the RG&E Seismic Safe Shutdown Equipment and Relay Review Lists for Unresolved Safety Issue (USI) A-46", which is part of the Ginna USI A-46 submittal, attached to the IPEEE submittal. This success path utilizes control rods and the Chemical and Volume Control System (CVCS) for reactivity control, the pressurizer heater and the power operated relief valves (PORVs) for RCS pressure control, the CVCS for RCS inventory control, and the Standby Auxiliary Feedwater (SAFW) system and the atmospheric relief valves (ARVs) for decay heat removal. This success path cannot handle a small LOCA case.

Although, as stated in the IPEEE submittal, that "all safety related component in the plant" are included in the assessment, a success path may not be available for the small LOCA case because some equipment that

are needed to handle a small LOCA may not be available under conditions of a review level earthquake (RLE). For example, some equipment associated with the safety injection (SI) system, the RHR system, and the CCW system (required for SI and RHR cooling) are on the Mechanical and Electrical Equipment Outliers list (Table 3 of the IPEEE submittal). Their reliability under an RLE was therefore questioned.

In response to an RAI on this issue, the licensee identified a second success path that includes the SI System for RCS reactivity and inventory control, the RHR system for RCS decay heat removal, and the CCW system as a support system for decay heat removal. This success path can handle a small LOCA. According to the response, all equipment and instrumentation necessary for the functioning of the second path was reviewed for seismic vulnerabilities, and all the outliers shown in Table 3 of the IPEEE submittal that could impact the functionality of the equipment selected for use in the second path have been resolved.

The second success path was also evaluated by the licensee for potential failure due to seismically indued damage from other equipment, and it is found to be vulnerable to failures caused by seismically induced flooding. The Reactor Makeup Water Tank and the Monitor Tank, if failed, can cause the interruption of one or more of the systems selected for the second success path. According to the licensee's response to the RAI, "these tanks will be considered outliers and will be examined to determine the correct course of action to reduce as needed the core damage risks associated with a seismic event."

According to the original IPEEE submittal and the licensee's response to the RAI, the equipment selected in the SSEL seems to be able to provide two success paths for safe shutdown following a seismic margin earthquake.

2.5 Plant Walkdown Approach

The IPEEE walkdown approach and the procedure utilized by Ginna in the walkdown process are the same as Ginna′s USI A-46 walkdowns, and follow the requirements of the Seismic Qualification Utility Group (SQUG) Generic Implementation Procedure (GIP). The submittal has addressed the seismic capacities of structures and safety components, and anchorage capacity evaluations, as well as interaction concerns and containment performance. Tables 2-3 and 2-4 of EPRI NP-6014 for focused-scope plants were used as screening criteria to prescreen structures and components.

A total of 908 items of mechanical and electrical equipment were reviewed based on the A-46/IPEEE walkdowns and the screening Tables 2-3 and 2-4 of EPRI NP-6041 for focused-scope plants. Seismic issues were identified for 52 items of equipment, which are listed in Table 3 of the submittal. Approximately 90 items of equipment were identified as being vulnerable to block walls, which are listed in Table 4 of the submittal. The walkdown approach appears to be appropriate.

2.6 Structural Analysis and High Confidence of Low Probability of Failure (HCLPF) Calculation

2.6.1 Structural Analysis and In-Structure Response Spectra

All Ginna's Seismic Category I structures are founded on rock. Based on the Screening Tables 2-3 and 2-4 of EPRI NP-6041 and the plant upgrades which resulted from SEP and other programs, all civil structures were screened at 0.3g.

The Ginna In-structure Response Spectra (ISRS), which were used for the IPEEE component evaluations, are the same as the spectra used for the USI A-46 evaluations. The ISRS were developed by Gilbert, based on the 0.2g RG 1.60 spectrum.

2.6.2 SSEL HCLPF Calculations

The screening of equipment seismic capacity is based on Table 2-4 of EPRI NP-6041 for focused-scope plants. If a component did not satisfy the screening requirements, its seismic evaluation was then performed based on the upgraded design basis. The exception are certain anchorage evaluations where the anchored equipment is rigid. There the calculated factor of safety was reduced by 1.5 to meet the 0.3g screening requirement. To substantiate the component capacity evaluation, the licensee in their RAI response (Reference 2), provided lists of components that were evaluated either as rigid or in the frequency range over 10Hz, showing that these components have a factor of safety of 1.5 or greater and, therefore, meet the IPEEE screening criteria.

2.7 Soil Evaluation

Ginna is a rock site. Therefore, a soil evaluation is not needed.

2.8 Relay Chatter Evaluation

Relay chatter review was not performed in the seismic IPEEE because no low ruggedness relays were found at Ginna during the resolution of the USI A-46 relays.

2.9 Containment Performance

Containment performance is discussed briefly in Section 6 of the submittal. According to the submittal, a review of containment integrity was performed. The purpose of the review was to identify any vulnerabilities associated with early containment failure due to the postulated seismic event. None were identified.

The items evaluated include containment structure integrity, isolation systems, bypass systems, and plant unique containment systems. All safety-related, air, motor, and solenoid operated valves were reviewed. The design of the electrical and mechanical penetrations, the personnel and equipment hatches, and the fuel transfer canal were also reviewed. Piping penetrations that require allowance for differential displacement are equipped with a bellows type expansion joint, and no active isolation systems are utilized. The review did not identify any features which would give rise to an early containment failure concern.

It seems that important containment issues, as those raised in NUREG-1407, have been addressed in the IPEEE.

2.10 Nonseismic Failures and Human Actions

Non-seismic failures and human actions are not discussed in the IPEEE submittal, they are discussed in the licensee’s response to the IPEEE RAI and the Ginna USI A-46 report.

Non-seismic failures are discussed in the response in a very general sense. It is argued that the issue is addressed by the redundant components in the systems selected in the success paths. Furthermore, all of the equipment relied upon is the normal equipment set used in the plant emergency operating procedures. As such all of the appropriate non-seismic failures and human actions have been evaluated in the Probabilistic Safety Assessment or the SQUG assessment.

As mentioned above, operator actions involved in the two success paths are discussed in the licensee’s response to the RAI and the Ginna USI A-46 report. The effects of the potentially adverse environmental conditions during a seismic event on operator actions are discussed in RG&E’s June 7, 1999 response to a SQUG RAI, which is attached to the licensee’s response to the IPEEE RAI (Reference 2).

2.11 Seismic-Induced Fires/Floods

The potential for seismic-induced floods was evaluated by the SRT. The SRT identified the number of tanks in the auxiliary building as a potential concern. Subsequently, this issue was addressed and it was concluded that these tanks will not cause seismic-induced flooding.

Issues related to seismic-induced fire, seismic actuation of fire suppression systems, and seismic degradation of fire suppression systems were examined thoroughly by the SRT during the seismic capability walkdown. The following problems (the licensee used the term "vulnerability") were identified:

- The house heating boiler, which is located near the service water pumps in the screenhouse, was not anchored. It could shift and damage the attached natural gas line.

- There are several locations where block wall failures could result in the release of combustibles: an oxygen line in the auxiliary building, a hydrogen line and valve station in the intermediate building, and hydrogen cylinders in the turbine building.

- There are two fire suppression systems that could be actuated by block wall failures: the manual deluge system in the relay room, and both a manual deluge system and a pre-action sprinkler system on elevation 253 in the intermediate building.

- Block walls are used as fire barriers throughout the plant. The walls whose failure could impact the fire protection of safety related equipment are those separating the service building from the intermediate building (column line 3), and those separating the turbine building from intermediate building (column line F).

- The two reactor coolant pump oil collecting tanks in the containment basement were not reviewed during the seismic walkdown because the containment was inaccessible.

These issues were later resolved as a part of the Ginna's IPEEE Fire Analysis by either design evaluations or design changes. Detailed discussions are provided in the Ginna's IPEEE fire analysis submittal, Section 3.9.3.

2.12 Unresolved Safety Issues (USIs) and Generic Safety Issues (GSIs)

USI A-45 Shutdown Decay Heat Removal Requirements

USI A-45 was addressed by the inclusion of the decay heat removal systems and associated components in the IPEEE analysis.

GSI-131 Potential Seismic Interaction Involving the Movable In-Core Flux Mapping System Used in Westinghouse Plants

The Movable In-core Flux Mapping System was examined by the A-46/IPEEE SRT during the containment walkdown, and was found not to be seismically vulnerable.

GSI-156 Systematic Evaluation Program

Since Ginna is a rock site, all soil related issues do not apply. The seismic design of structures, systems, and components were addressed in the submittal with respect to ground response spectra, and in-structure response spectra. The seismic input used in the IPEEE evaluations was based on a 0.2g Reg. Guide 1.60 spectrum.

GSI-172 Multiple System Response Program

GSI-172 issues were addressed in the IPEEE submittal as follows:

- Effect of fire protection system actuation on safety-related equipment was addressed in Section 5.6.2 of the submittal and Section 3.9.3 of the Ginna IPEEE Fire Analysis.

- Seismic/fire interactions were addressed in Section 5.6.2 of the submittal and Section 3.9.3 of the Ginna IPEEE Fire Analysis.

- Hydrogen line ruptures were addressed in Section 5.6.2 of the submittal and Section 3.9.3 of the Ginna IPEEE Fire Analysis.

- Seismic-induced flooding was addressed in Section 5.6 of the submittal.

- Seismic-induced spatial and functional interactions were addressed in Section 5 of the submittal.

- Seismic-induced relay chatter evaluation was addressed in Section 5.4 of the submittal.

- Failures related to human errors were discussed in the RAI response (Reference 2).

2.13 Vulnerabilities/Plant Improvements

The term vulnerability is not defined in the submittal. The submittal does talk about identifying and assessing vulnerabilities. Section 5.6.2 of the submittal, Seismic/Fire Interaction, mentions that a number of "vulnerabilities" were identified in this area of the examination, as noted in Section 2.11 of this TER, and all were addressed.

The IPEEE seismic evaluation resulted in 52 items of equipment that could not be screened out, and were provided in Table 3 of the IPEEE seismic evaluation submittal; ductwork and post accident charcoal filter units in the containment building could not be screened out. According to the submittal, these components meet their existing licensing basis, and no further work will be performed by RG&E with respect to seismic issues outside of those related to USI A-46 closeout.

However, the licensee's response to the RAI on the selection of a second success path mentions an outlier which is still being investigated. As discussed in Section 2.4 of this TER, a second success path, for small LOCA, was evaluated by the licensee for potential failure due to seismically indued damage from other equipment, and it was found to be vulnerable to failures caused by seismically induced flooding. The Reactor Makeup Water Tank and the Monitor Tank, if failed, can cause the interruption of one or more of the systems selected for the second success path. According to the licensee's response to the RAI, "these tanks will be considered outliers and will be examined to determine the correct course of action to reduce as needed the core damage risks associated with a seismic event."

3.0 OVERALL EVALUATION AND CONCLUSIONS

Although Ginna, which is a focused-scope plant, performed a reduced-scope seismic evaluation in the original IPEEE submittal, the licensee's RAI's response has provided additional information and discussions to address the issues that are required for a focused-scope plant. In light of both the IPEEE submittal and the RAI response, the overall process, methods, and organization of the licensee's IPEEE program appear satisfactory with respect to the requested information outlined in NUREG-1407. Ginna listed all safety related structures, components, and performed the seismic evaluations based on the Screening Tables 2-3 and 2-4 of EPRI NP-6041 for focused-scope plants and the 0.2g RG 1.60 spectrum.

The submittal addressed all the GSIs as requested in NUREG-1407.

Based on our evaluations, it appears that the licensee has satisfied the objectives outlined in the Generic Letter with respect to the IPEEE program.

4.0 REFERENCES

[1] IPEEE Seismic Evaluation Report (prepared by Stevenson & Associates), January 1997, Rochester Gas and Electric, Robert E. Ginna Station, Attachment to Letter dated January 31, 1997 from Robert C. Mecredy, Vice President, Nuclear Operations, Rochester Gas and Electric Corporation, to USNRC.

[2] Response to the NRC's Request for Additional Information, Attachment to Letter dated July 30, 1999 from Robert C. Mecredy, Vice President, Nuclear Operations, Rochester Gas and Electric Corporation, to USNRC.

Attachment 2

R.E. GINNA NUCLEAR POWER PLANT

INDIVIDUAL PLANT EXAMINATION OF EXTERNAL EVENTS (IPEEE)

TECHNICAL EVALUATION REPORT

FIRES

Review of the Submittal in Response to U.S. NRC Generic Letter 88-20, Supplement 4: "Individual Plant Examination-External Events"

Fire Submittal Screening Review Technical Evaluation Report: R. E. Ginna Revision 2: October 13, 1999

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1.0 INTRODUCTION

This Technical Evaluation Report (TER) presents the results of the Step 0 review of the Ginna fire assessment reported in “R.E. Ginna Nuclear Plant IPEEE Fire Analysis” [1], requests for additional information (RAI) based on questions raised during the initial review [2], and the licensee responses to those questions [3].

1.1 Plant Description

Ginna is a Westinghouse two-loop pressurized water reactor (PWR) with a power rating of 1520 MW_t. The nuclear steam supply system (NSSS) is similar to other Westinghouse plants and includes an emergency core cooling system (ECCS) consisting of three high-pressure safety injection (HPSI) pumps, a low-pressure safety injection (LPSI) system which is an operating mode of the residual heat removal (RHR) system, and two accumulators. Two of the HPSI pumps each provide flow into the vessel through separate cold legs of the reactor coolant system (RCS). The third HPSI pump can provide flow to the RCS through either of the cold legs. The LPSI system has two pumps each of which can provide flow to either of the RCS cold legs. The refueling water storage tank is the water source for the HPSI and LPSI during the injection mode of ECCS operation. The LPSI (RHR) pumps are used in the recirculation mode of ECCS operation providing flow from the containment sump to the HPSI pump suction or to the vessel directly. The RHR pumps also provide the shutdown cooling function. Decay heat removal capability is provided by the main feedwater system which has two pump trains, a main auxiliary feedwater (AFW) system consisting of two motor-driven and one steam-driven pumps, and a standby AFW system consisting of two motor-driven pumps; by a feed and bleed operation; and by the RHR system. A separate containment spray system (CSS) consisting of two pumps is also available. Similar to the HPSI, the CSS takes suction from the RWST during the injection mode of operation and switches suction to the RHR pumps during the recirculation mode of operation. Emergency ac power is provided by two divisions of emergency power each powered by a separate diesel generator. Component cooling is provided by the component cooling water (CCW) system (cools the RHR heat exchangers and other emergency loads) and the service water system (cools the diesel generators, the CCW heat exchangers, and the HPSI pumps).

The major structures at Ginna are the containment, the auxiliary building, the intermediate building, a diesel generator building, the service building, a control building, a screen house, and the turbine building. The RCS and portions of the ECCS, AFW and RHR systems are located within the containment. The auxiliary building contains major components of the ECCS, CCW, RHR, and electric power system. The intermediate building contains the main AFW system. The control building houses the control room, relay room, and battery rooms. The standby AFW system is located in a separate standby AFW building. The diesel generator building contains the two diesel generators and their associated electric boards. The screen house contains the service water pumps and the fire water pumps. The turbine building contains the components of the power conversion system.

The licensee listed the following plant safety features as providing significant reduction in fire risk at Ginna:

- Automatic fire suppression systems installed in the relay room, cable tunnel, and diesel generator room B.

- Sprinklers installed in cable trays located in the auxiliary building basement and mezzanine, as well as the intermediate building basement and the air handling room.

- Locked open circuit breakers associated with certain MOVs installed in the plant to significantly reduce the likelihood for spurious transfers due to hot shorts.

- Fire wrap installed on cables associated with charging pump A.

- The standby AFW system which adds significant redundancy and diversity to the AFW and main feedwater systems.

1.2 Review Objectives

The performance of an IPEEE was requested of all commercial U.S. nuclear power plants by the U.S. Nuclear Regulatory Commission (USNRC) in Supplement 4 of Generic Letter 88-20 [4]. Additional guidance on the intent and scope of the IPEEE process was provided in NUREG-1407 [5]. The objective of this Step 0 screening review is to help the USNRC determine if the Ginna submittal has met the intent of the generic letter and to also determine the extent to which the fire assessment addresses certain other specific issues and ongoing programs.

1.3 Scope and Limitations

The Step 0 review was limited to the material presented in the Ginna IPEEE submittal and responses to requests for additional information (RAI). The scope of the review was limited to verifying that the critical elements of an acceptable fire analysis have been presented. An in-depth evaluation of the various inputs, assumptions, and calculations was not performed. The review was performed according to the guidance presented in Reference 6. RAIs were submitted to the licensee based on an initial review of the submittal alone. The results of the review of the submittal and RAI responses against the guidance are presented in Section 2.0. Conclusions and a recommendation as to the adequacy of the Ginna IPEEE submittal with regard to the fire assessment and its use in supporting the resolution of other issues are presented in Section 3.0.

2.0 FIRE ASSESSMENT EVALUATION

The following subsections provide the results of the review of the Ginna fire assessment. The review compares the fire assessment against the requirements for performing the IPEEE and its use in addressing other issues. Both areas of potential weakness and strength of the fire assessment are highlighted.

2.1 Compliance with NRC IPEEE Guidelines

The USNRC guidelines for performance of the IPEEE fire analysis derive from two major documents. The first is NUREG-1407 [5], and the second is Supplement 4 to USNRC Generic Letter 88-20 [4]. Factors for determining the adequacy of a fire assessment in comparison to these guidelines, as determined in a Step 0 review, have been outlined in "Guidance for the Performance of Screening Review of Submittals In Response to U. S. NRC Generic Letter 88-20, Supplement 4: Individual Plant Examinations - External Events," Revision 3, March 21, 1997 [6]. The following sections discuss the utility document in the context of the specific review objectives set forth in this screening review guidance document and assesses the extent to which the utility submittal has achieved the stated objectives.

2.1.1 Methodology Documentation

The Ginna fire assessment utilizes a probabilistic risk assessment (PRA) to systematically and successively evaluate fire hazards and their associated risks. The analysis was performed in three phases. The licensee states that the first two phases, consisting of qualitative and quantitative screening steps, utilize methods which are consistent with the Fire-Induced Vulnerability Evaluation (FIVE) [7] methodology which was approved for use in NUREG-1407 for screening. The third phase is a detailed fire PRA which was performed for fire areas and fire zones that were not screened. In Phase 1, qualitative screening of buildings and structures was performed according to specified criteria which are consistent with the criteria used in FIVE for screening fire areas. In Phase 2, remaining buildings were divided into fire areas and fire zones which were then quantitatively screened using the same approach specified in FIVE (i.e., the core damage frequency due to fire in each location was evaluated assuming that all equipment and cables in a location were damaged by any fire). Phase 3 of the Ginna assessment is a detailed fire evaluation of the remaining unscreened fire areas and compartments. In this phase, fire scenarios were developed and quantified for each location, depending on the physical configuration of the compartment, the availability of fire suppression systems, and the location of combustibles and equipment. Fire scenarios that propagate from one location to another were considered in this phase. Phase 2 and 3 of the analysis used models developed in the plant's internal events PRA. The major assumptions which were utilized in the fire assessment and their validity are listed below. Additional assumptions are discussed in the remainder of this report.

- The fire frequency associated with components were obtained by using generic fire data [8] and actual Ginna experience using a two-stage Bayesian update technique. This is viewed as a positive aspect of the analysis.

- The quantitative portion of the Ginna fire assessment used the internal events PRA models which apparently includes non-Appendix R equipment. The locations of this equipment and associated cables were identified and used in the assessment of fire damage. Giving credit for both Appendix R and non-Appendix R equipment is viewed as a positive aspect of the Ginna fire assessment.

- The potential for cable hot-short induced failures was modeled explicitly in the PRA portion of the analysis. Probabilities for hot shorts were assigned for different components and ranged from 1E-3 to 0.1. The inclusion of cable hot shorts in the fire PRA is viewed as a positive aspect of the analysis.

- Cable wrap was credited with protecting cables from fire damage. A probability of 0.15 was assigned to the failure of cable wrap which the licensee equates to the probability that a fire is not suppressed within the one hour duration of the wrap's fire rating. Crediting a cable wrap in this fashion could lead to double counting suppression if suppression is modeled in areas containing the wrap. (An RAI response indicated that double counting was avoided.)

- No fire modeling was performed to support the quantitative fire assessment. Instead, levels of damage were assumed based on knowledge of the contents of each location. In general, unsuppressed fires in the Phase 3 detailed fire PRA evaluation were assumed to destroy all the equipment and cables in the location. The damage associated with suppressed fires was more variable, but in many cases the loss of an entire ac division was assumed.

- In the Phase 3 detailed fire PRA evaluation, the probability that a fire inside an electrical cabinet will damage equipment outside of the cabinet due to flame propagation or development of sufficient heat was assigned a value of 0.1. No fire modeling was performed for electrical cabinets. The licensee justifies this probability based upon the fact that the cabinets at Ginna are completely enclosed and generally compartmentalized and upon a review of the electrical cabinet fires in the EPRI fire events database [9]. Although the use of this 0.1 severity factor may be a reasonable estimate of the probability of exterior damage caused from fires in enclosed electrical cabinets, the appropriateness of this factor to all electrical cabinet fires in Ginna cannot be confirmed from the submittal. It is always preferred that scenario-specific fire modeling that considers the contents of the cabinets and the spatial location of cables above the cabinet be performed.

The methodology description of the original submittal was lacking in some regards. For example, although stated as in accordance with the FIVE prescription, the conduct of the fire compartment interaction analysis (FCIA) could not be verified. In this particular

case, an RAI was submitted to the licensee and the response is described below. In other cases, the presentation of the results or the discussions in response to other RAIs provided insights as to the methods employed. Taken together, the submittal, RAI responses, and results presentations give a clear picture of the methodology.

2.1.2 Plant Walkdown

The submittal indicates that the information used in the Ginna fire assessment included the PRA model, general arrangement drawings the Updated Final Safety Analysis Report, the Appendix R analysis, a cable routing database and cable routing drawings, plant records of fire incidents, and other sources. (These documents were not included with the IPEEE fire submittal.) Relevant information for the analysis was stored in a relational spatial interactions analysis database. A set of Location Characteristic Tables (LCTs) was developed to summarize the stored information. Information in the LCTs was verified during plant walkdowns and continuously updated throughout the analysis.

The Ginna submittal indicates that two plant walkdowns were performed in support of the fire assessment. A brief description of the walkdown teams was provided indicating that two plant engineers and two contract personnel made up the walkdown team. The credentials of the walkdown team were not provided. The submittal does not indicate whether plant procedures were used to guide the walkdown effort.

The first walkdown was conducted at the beginning of the quantitative screening. The purpose of this walkdown was to gain an early appreciation of the spatial interactions between fire sources and equipment, to confirm information that had been gathered in the LCTs, to inspect the amount and location of transient hazards, and to identify potential fire propagation pathways. For plant areas that could not be visited due to radiological controls, a digital photo database was used to collect and verify information. Walkdown notes were added to the LCTs.

The second walkdown was conducted at the beginning of the detailed PRA assessment of specific fire scenarios. The purpose of this walkdown was to confirm the results of the quantitative screening and, for the remaining unscreened fire zones, to identify the exact locations of safety-related equipment and cables, inspect the amount and location of transient fire hazards, and develop scenarios for the detailed fire assessment.

The walkdown findings are not presented in the submittal but are noted as being described in a second tier document. Overall, the walkdowns as described in the submittal appear to cover the main requirements for acceptance, including the collection of data as well as the confirmation and verification of data. Although a walkdown was performed to confirm the results of the quantitative screening, it is not apparent from the submittal that a confirmation of the qualitative screening was obtained through a walkdown.

2.1.3 Qualitative Screening

Qualitative screening was performed on the basis of the absence of safety-related equipment combined with the combustible material loading in a plant location. A plant location was considered to be functionally important if it contains fire-susceptible safety-related plant equipment and/or its associated power, control and instrumentation cables, whose damage could cause an initiating event or interfere with the plant's ability to mitigate accidents. This is equivalent to the FIVE criteria for screening fire areas. An additional consideration used to determine the importance of a location was whether it contains a sufficient amount of combustible material that, if ignited, could result in a fire that could potentially damage safety-related equipment or propagate to adjacent zones containing fire-susceptible safety-related equipment or cabling. No specific criteria for determining when a fire will propagate between locations were listed in the Ginna submittal. (An RAI addressing this concern is addressed below.)

The above approach was used in a preliminary screening performed at the building/structure level. As a result of this screening, 10 buildings/structures were eliminated from further analysis. A review indicates that none of the eliminated buildings/structures would be expected to be significant risk contributors.

The plant buildings/structures retained for further analysis were subdivided into smaller areas for the subsequent analyses. The submittal states that the fire areas and fire zones defined in the Ginna Fire Hazard Analysis (FHA) performed for the Appendix R report were used as a basis for the rest of the study. A fire area is defined as a location surrounded by a 3-hour or 2-hour rated barrier. A fire zone is defined in the submittal as a subdivision of a fire area which may consist of a compartment or several compartments, or it may be a fire area itself. The submittal states that there is no barrier rating requirement for fire zones. In general plant locations were grouped together into a fire zone based on the following criteria:

- fires can propagate freely within the locations,
- the locations contain equipment whose failures cause similar impacts,
- the locations are protected by similar fire protection capability,
- fire hazard contents are similar in those locations,
- a physical barrier separates the locations from the rest of the plant areas and there is a significant time delay for fire propagation from the zone to other locations.

Section 3.1 of the submittal indicates that fire areas were qualitativly screened according to the FIVE criteria, i.e., the area contains no Appendix R equipment and a fire in the area would not cause a demand for a safe shutdown. Furthermore, this section of the submittal indicates that fire compartments (not zones) were screened if they met the above criteria and had no credible potential for fire spreading to other fire compartments. The criterion for determining when fire propagation is assumed to occur is not specified in the submittal.

The last bullet listed above implies that physical barriers (not necessarily fire-rated) were assumed to prevent propagation. As stated, this criterion does not conform to the FIVE method, which addresses non-combustible barriers in combination with low combustible loads and fire detection and suppression.

Whether or not specific fire areas and fire zones (or compartments) within the remaining unscreened buildings were qualitatively screened is not clear from the information provided in the submittal. Table 3-2 of the submittal lists 50 fire zones that were not qualitatively screened in the analysis. The qualitative screening did not eliminate buildings that are typically important from a fire risk standpoint.

In summary, the approach used for qualitative screening of buildings utilized criteria similar to those used in FIVE for screening fire compartments. It is unclear, however, whether further qualitative screening, such as an FCIA, was performed in the fire assessment. The main difference is that the criteria for determining if a fire can propagate from one location to another was not specified. To resolve this ambiguity, and RAI was submitted to the licensee.

The licensee response to this RAI (RAI #1)included excerpts from two appendices produced during the study, but not a part of the IPEEE submittal. The conclusion from the study was that there were no credible propagation pathways that resulted in multi-compartment fires. A variety of arguments were offered in support of this conclusion.

Regarding seals and dampers, the response noted that these are regularly inspected and tested. In summarizing its response, the licensee stated that the effects of fire, heat, and smoke propagation had been addressed. However, only fire propagation appears to have been explicitly addressed. Also, the response noted the use of the following correlation between the combustible load in a compartment and the fire duration:

$$\text{Duration(hour)} = \text{combustible load/ 80,000 Btu.}$$

This duration was then used with barrier ratings to determine whether the barrier would be modeled as failed, as described below.

In its response, the licensee noted that the potential for fire propagation considered two criteria:

- The existence of a permanent opening between compartments,
- A combustible load that exceeds (by the correlation above) 75% of the enclosing barrier rating (hours).

Compartments assessed as having a propagation concern were further investigated. One observation was that, because of propagation delays from barriers, for example, only compartments adjacent to the compartment containing the fire needed to be considered. Various suppression means were credited in making this conclusion. The

licensee stated that the 25% barrier derating introduced additional conservatism into the evaluation and accounted for such concerns as a normally-closed door being left open. (This last point was not supported by any argument.)

The tabulations provided indicate that a thorough assessment according to these criteria was made. This assessment included the high hazard areas that the RAI specifically requested be addressed. However, for all barriers assessed as having a potential propagation concern, additional criteria were offered, eventually leading to the conclusion that no propagation cases were of concern. Among these additional criteria were:

- Concrete walls and steel hatches were credited.
- Unrated barriers were credited when physically separated from combustibles.
- Concrete curbs were credited with containing collected oil spills.
- Separation and installed suppression systems were credited with precluding particular propagation paths.
- Limited air flow was credited with preventing large fires.
- Unrated barriers were credited when combustible loads were small.

The argument for derating barriers (equivalence to open doors) was not developed beyond the statements noted. Also, no basis for the correlation between combustible load and fire duration was given. However, it is not clear that either of these assumptions has a serious impact on the screening result. The licensee has provided a clear response to the question.

2.1.4 Fire Occurrence Frequency

In the next phase of the Ginna fire assessment (Phase II of the FIVE methodology), quantitative screening was performed for the remaining fire zones based on the 1E-6/yr core damage criteria allowed in the FIVE methodology. Performance of quantitative screening required generation of fire initiation frequencies for each fire zone. Generic fire data contained in a database generated by Pickard, Lowe, and Garrick was reviewed for applicability to Ginna. Events that could not occur at Ginna (e.g., fires in components not present at the plant) and fires that can only occur during shutdown were removed from the database. A two-stage Bayesian analysis was then performed to combine the industry data with actual Ginna fire data. Plant-specific fire event data was compiled from Fire Brigade reports that covered the period from 1979 to 1997.

The licensee indicates that three types of fire event categories were defined and used in evaluating fire frequencies. The first is location-specific fire events that generated frequencies for the containment, auxiliary building, turbine building, and screen house. The licensee indicates that it is appropriate to use location-specific fire frequencies for large areas that contain a variety of mechanical equipment. The second type of fire event results in frequencies specific to the equipment present. The licensee states that this data should be used for locations which contain a single type of component, unique components that represent a fire hazard, or for locations containing electrical equipment

including cabinets and cables. A list of 16 components that can result in a fire is provided in the submittal. Missing from this list are pumps and air compressors. The third type of fire event listed by the licensee is the control room fire, which was addressed separately due to the unique nature of the control room.

The licensee indicates that a fire frequency for a fire zone was obtained by apportioning the total fire frequency for each component by the ratio of the number of components contained in the fire zone to the total number of that component at the plant. According to the submittal, the fire frequencies reflect the variety and number of components, in-situ fuel sources, fuel loading, floor area, and personnel activities within each fire zone. The licensee states that during the development of the fire frequencies, transient combustible fires were grouped with the type of component that was primarily damaged by or exposed to the fire. Based on this brief description, it is unclear how the fire frequencies were apportioned. However, an RAI response provides a further description of this process.

No information was provided in the submittal to check the method and results of the apportionment of the fire frequencies (details are referenced as being contained in a tier 2 document). The resulting frequencies are presented in the submittal and a comparison of the calculated fire frequencies with information in NUREG/CR-4840 [10] and FIVE [7] indicates that the calculated fire frequencies are in general agreement with these references. Fire frequencies are presented for 48 fire zones.

Two fire zones, IBN-0 and IBS-0 which are in the intermediate building sub-basement and which were not qualitatively screened, were not subjected to quantitative analysis for reasons unspecified in the submittal. The missing information on zones IBN-0 and IBS-0 resulted in an RAI (RAI#5) to the licensee which, unfortunately, was misworded to ask how zones IBN-1 and IBS-1 were treated. (The licensee responded with a list of submittal citations describing the disposition of zones IBN-1 and IBS-1.) NRC Project Manager, Guy Vissing, provided the information obtained from the licensee indicating that (1) IBS–0 contains no safety equipment, and (2) IBN-0 contains only lube oil piping for the turbine-driven auxiliary feedwater pump. Thus, qualitative screening of both areas was reasonable.

2.1.5 Quantitative Screening

Forty-eight of the unscreened fire zones remaining at this point in the fire assessment were subjected to a quantitative screening analysis. The quantitative screening appears to have followed the FIVE methodology. All equipment located in a fire zone was assumed to be failed from any fires in the compartment. The Ginna PRA was used to calculate the conditional core damage probability (CCDP) arising from fire damage in each fire zone. For the fire zones that were quantitatively screened, the CCDPs ranged from 5.3E-7 to 8.2E-4. These probabilities were multiplied by the fire frequency calculated for the corresponding compartment to obtain screening estimates of the core

damage frequencies. Fire detection and suppression were not credited in the quantitative screening process.

Details of the PRA model used in the fire assessment are not provided in the submittal. However, discussion of some of the fire scenarios presented in the submittal indicates that loss-of-coolant accidents (hot-short induced opening of pressurizer PORVs) and loss-of-offsite power (LOSP) accidents in addition to plant trips and loss of feedwater transients were included in the quantitative evaluations. The impact of the fire on post-initiator human errors contained in the PRA models was also addressed in some detail in the submittal. In many cases, the PRA values for the human error probabilities (HEPs) were used in the quantitative assessment. Justification for this was usually the fact that damage to indicating circuits from fires was not identified in the analysis (implying that instrumentation circuit cables were traced). In some cases, the HEP for an event was unchanged since the licensee states that the fire impact on instrumentation was manifested in another HEP. This implies dependency between the HEPs which the licensee states was accounted for in the analysis. Finally, some HEPS were unchanged since there is a substantial amount of time to perform the required action.

For many of the HEPs, the values used in the internal events PRA model were replaced with screening values that ranged from 0.01 to 0.5. The screening values are higher than the HEPs used in the PRA and account for increased stress caused by the fire. Local actions were credited if significant time would be available for heat and smoke removal before the action is required. No local operation of equipment (e.g., manually operating a motor-operated valve) was allowed in the analysis documented in the submittal. (Manual actions were credited in the response to an RAI.) HEPs were also increased if instrumentation cables related to the action were not traced. The licensee states that the higher HEP was assumed to account for the potential loss of indication. Higher HEPs were also assigned if direct indication would be lost but alternate indications would be available to indicate that the action is required. From the description of the human error events included in the PRA model, there appears to some dependency between some of the events. The licensee states that these dependencies were handled in the quantitative analysis by raising the HEPs for the successive actions.

The submittal indicates that hot shorts were included in the quantitative portions of the assessment (it is not clear if this was done in both the screening and detailed analyses). This was accomplished by reviewing the Ginna internal events PRA models to identify components susceptible to hot shorts. The probabilities for basic events representing the failure of these components were replaced with assumed probabilities for hot shorts. Hot shorts of cables for solenoid valves were assumed to energize the valves or cause them to remain energized. A probability of 0.1 was assumed for hot shorts in air-operated valves (AOVs). Hot shorts of cables for motor-operated valves (MOVs) were assumed to cause the valves to transfer position. A probability of 0.1 was assigned for hot shorts occurring in MOV control cables. However, if a procedure is in place to open the control circuit for an MOV in the event of a fire (e.g., by removing a fuse), the

probability of a hot short was reduced to 0.03 since hot shorting of two cables would have to occur. The probability of a hot short in an MOV power cable was assumed to have a probability of 0.001 based on the low likelihood that the correct combination of multi-phase hot shorts on a spiral run power cable necessary to change the MOV position will occur.

Hot shorts of cables associated with circuit breakers were postulated to cause the breakers to spuriously transfer position. No probability for this type of hot short was specifically discussed in the submittal. The licensee states that hot shorts in cables associated with pumps, air compressors, and fans would not cause undesirable effects. Pump deadheading due to spurious starts or closure of outlet valves was found by the licensee to not contribute significantly to fire-related risk. However, the licensee indicates that hot shorts in the turbine-driven AFW pump control cables was accounted for (no probability was identified in the submittal) since pump overheating would occur if a hot short causes either steam admission valve to open before the pump discharge valves oven. Finally, the licensee states that hot shorts in cables for HVAC dampers were not modeled since some dampers would fail in a desirable state and since fire damage for other damper cables would also result in open circuits for the associated fans.

The submittal also indicates that cable wrap was credited in the quantitative assessments (it is not clear if this was done in both the screening and detailed analyses). A probability of 0.15 was assigned to the failure of cable wrap to account for the probability that a fire is not suppressed within the one hour time frame associated with the fire rating of the cable wrap. Modeling cable wrap failure may implicitly credit fire suppression in the quantitative screening of fire zones that credit cable wrap for protection. Crediting cable wrap when fire suppression is explicitly credited, can lead to double counting suppression efforts. An RAI was submitted to the licensee to ask for the details of how these credits were implemented.

The cable-wrap credit ambiguity resulted in an RAI (RAI#2) to the licensee. The licensee responded that the 0.15-factor was intended to indicate the probability of failure to manually suppress the fire prior to a 1-hour exposure of the cable wrap to the fire environment. Included in the quantification was the correlation relating the fire duration to the combustible load, discussed above. For cases with an insufficient combustible load, the assumption was made that 10% of the fires would have sufficient duration to challenge the cable wrap. Also, the response stated that failure of automatic suppression was explicitly modeled, which would imply that the automatic suppression reliabilities were included in the quantification.

The question of interpretation of the 0.15-factor is thus resolved.

The results of this quantitative screening step resulted in core damage frequencies <1E-6/yr and elimination of 29 of the remaining fire zones. Fire zones that were screened include the charging pump room, one of the diesel generator rooms (division B), the

standby AFW pump room, most of the fire zones in the turbine building, and all of the fire zones in the intermediate and service buildings. The remaining unscreened fire zones include the cable tunnel, both battery rooms, one of the diesel generator rooms, the main control room, the screen house, parts of the containment, most of the auxiliary building, and the transformer yard area.

In summary, the quantitative screening methodology used in the Ginna fire assessment appears to be reasonable. The estimate of the CDF contribution from the screened areas is 1.2E-6/yr.

2.1.6 Fire Propagation and Suppression Analysis

The 19 fire zones remaining after the qualitative and quantitative screening were subjected to further detailed evaluation including the analysis of fire propagation and suppression. Actual fire modeling using the FIVE methodology or other techniques was not performed. Instead, the licensee assumed probabilities for fire propagation primarily based upon physical separation of equipment. Propagation probabilities were assigned for a few fire scenarios in the turbine and auxiliary buildings. These probabilities range from 0.01 to 0.07. The auxiliary building basement (fire zone ABB) was further subdivided into three sections for the purpose of defining fire scenarios. Propagation between these locations was assumed by the licensee to not be credible due to either the presence of concrete walls or adequate separation (i.e., 40 feet) and the presence of the RWST in the intervening space.

For the detailed fire PRA evaluation performed in Phase 3 of the assessment, a 0.1 value was assigned as the probability that a fire inside a cabinet will damage equipment outside of the cabinet due to the development of sufficient heat or the development of flames outside the cabinet. The licensee justifies this probability by noting that the cabinets to which this value was applied in the analysis are completely enclosed and typically compartmentalized. This configuration is stated as limiting oxygen availability, likely to reduce the amount of heat released in the fire, and increasing the time required for the fire to grow substantially. In addition, the licensee states that only one of the over 100 cabinet fires in the EPRI Fire Events Database [9] appears to have damage equipment outside the cabinet. The licensee further states this probability is consistent with electrical cabinet fire tests performed by Sandia National Laboratories where all of the test fires self-extinguished without propagating to adjacent cabinets (a tier 2 IPEEE document was referenced in the submittal).

According to the submittal, transient combustible fires were not analyzed separately. The licensee states that during the development of the fire frequencies, transient combustibles were grouped with the type of component that was primarily damaged by or exposed to the fire. Thus, the licensee states that the impact and consequences of transient combustibles fires are accounted for in the modeled component fires and no separate evaluation of transient fires was necessary. The details on how this assumption was implemented were the subject of an RAI.

RAI #4 addressed the treatment of transient combustible materials fires in the submittal. The submittal stated that the transient fire frequency was included in the components likely to be damaged from such fires. The details of how this was done were not clear. In particular, it was not clear how frequencies associated with fire damage to equipment were allocated.

The response indicates a general treatment of frequencies that were calculated for an entire area in a manner similar to FIVE, including the transient fire contribution. Thus, compartment fire frequencies were calculated in a manner that included transient ignition sources and combustibles.

The response to RAI #4 did not elaborate on the question of assignment of a conditional probability of component damage. However, the response to RAI #3 (below) indicates that damage frequency was allocated as a fraction of the compartment fire frequency. That fraction was based on the layout of components within the compartment and, in one example case, on fractional floor area.

Based on the response, it appears that the question of the treatment of transient combustible materials fires has been subsumed by the general question of the treatment of fire damage. This question is addressed by RAI #3 below.

From the description of fire scenarios presented in the submittal, it appears that fires that are not suppressed were assumed to damage all the equipment in the fire zone. The licensee indicates that it is likely that all such fires would not result in complete damage. It is not known if this assumption was used in all fire zones. If the fire was suppressed, some level of damage was assumed to occur. It appears that in many cases, suppressed fires were assumed to damage one electrical division. The licensee indicates this is conservative since for many fire scenarios only a portion of components relying on the electrical division would be disabled. It is noted that some suppressed fires were assumed to cause a reactor trip, a reactor coolant pump seal LOCA, a loss of feedwater, or a LOSP.

As stated earlier, the licensee indicated that fire suppression was not credited in the quantitative screening assessment. However, fire suppression was credited in the detailed fire PRA analysis. Both automatic and manual fire suppression were credited in some cases. The failure of automatic fire suppression was assigned a generic failure probability of 0.05 (used for deluge, pre-action, and Halon systems). The automatic suppression systems were assumed by the licensee to comply with the National Fire Protection Association (NFPA) and NRC guidelines. The fire protection systems were assumed to be maintained regularly. The verification of NFPA compliance was the subject of an RAI.

The submittal stated that a high reliability was assumed for fire suppression systems. RAI #7 asked that the compliance of the fire suppression systems with standards, such

as those of NFPA, be verified. The assumption of highly reliable suppression systems is generally justified if based on such compliance.

In response, the licensee noted that its original suppression systems were installed in accordance with standards applicable in the late 1960's. Subsequent upgrades and improvements, notably those associated with the Systematic Evaluation Program and Appendix R, used NFPA standards as guidance. The response noted that the systems' designs, and their maintenance and testing, are all in accordance with plans approved by NRC.

The response noted that the fire suppression systems are in compliance with regulations, but did not provide a basis for the reliability values assumed, which is a weakness of the study. While the question addressed risk, the response described regulatory compliance. The potential exists for the introduction of optimism into the results from assuming highly reliable fire suppression systems.

Manual fire suppression was credited for selected fire scenarios. Probabilities of non-suppression were assigned based upon reviews of plant fire drill records and events in the EPRI Fire Events Database [9]. The latter were reviewed specifically with respect to the type of suppression, time to suppress, and severity of damage to nearby equipment. In addition, for each fire scenario in the fire PRA, the location of manual pull stations was determined and access routes were assessed. Based upon these evaluations, a probability of 0.1 or 0.33 was assigned for failure to manually actuate automatic suppression systems in selected areas given the failure of the system to automatically actuate. (These values were also discussed in an RAI response.) Failure to manually suppress a fire in several locations using extinguishers or hose stations was assigned a failure probability of 0.33. Manual suppression failure for control room fires was assigned a probability of 0.1 (the licensee notes that this is conservative compared to the values quoted in NSAC-181 [11]).

No criteria were provided in the submittal for determining when fire propagation between fire zones would occur. In fact, only a few propagation scenarios were identified. Those identified utilized subjectively assigned propagation probabilities without providing their bases. The level of damage caused by suppressed fires appears to have been assumed by the licensee. Although it appears to have been done by taking into account the spatial layout of the fire zones, it cannot be verified that the assumed damage levels are realistic or bounding. It is noted that one fire scenario described in the submittal credited automatic fire suppression, manually actuated suppression, and manual fire suppression. It is unclear if dependencies between the failure of these events were properly modeled. Two RAIs resulted from the treatment of suppression, one addressing explicit modeling of propagation versus suppression, and another addressing the dependencies between the various modes of suppression.

The study documented in the submittal credited combinations of fire suppression methods, including automatic suppression systems, manual suppression, and manual

actuation of installed suppression systems. RAI #9 questioned the assumption of independence of all of these suppression methods.

In response, the licensee stated that no significant dependencies were found during its study. The response noted that three independent water supplies were available. It further noted that manually actuated systems require actuation from the control room, so they are believed to be independent of manual suppression efforts. Finally, dependencies between manually actuated systems and automatic systems were addressed in the plant model by allowing only the recovery of failures in actuation circuits.

The licensee has responded appropriately to the RAI.

Throughout the submittal, the treatment of fire propagation versus fire suppression, the extent of damage to particular pieces of equipment, the timing of damage, and assumed limits to damage were not well explained. RAI #3 was directed at resolving this ambiguity. The licensee response provided information on the reasoning used without providing information on or the basis for, the probabilities used in quantification.

The response states that, in general, fires were assumed to fail all equipment and circuitry within a fire compartment unless a specific argument could be made that the extent of damage was limited. Furthermore, it appears that damage consequences were usually assumed to be bounded by some known conditional core damage probability (CCDP), most frequently that of the loss of a single electrical division. Arguments for limited damage were made in terms of suppression success (in which case the loss of one electrical division was assumed), or physical separation and barriers. For the latter case, specific equipment damaged and frequency partitioning were identified for each compartment. The number of these cases is small. The response is broken into two subsections, described in the following paragraphs. The discussion also provides insight as to how fire damage was assessed.

(A) Damage limited to one division in cases where suppression succeeds: Cables and equipment were treated separately. For cabling, the particular systems represented in the cable inventory of each tray resulted in system losses with a CCDP that was bounded by the loss of a single electrical division. Notably, the response stated that manual actuation of sprinkler systems was assumed to fail only 1% of the time. This fraction was given as 10% in the submittal.

For equipment, five zones were identified as being of particular interest since they contained multiple equipment trains. These were the auxiliary building basement (zone ABB), the auxiliary building mezzanine (zone ABM), the air handling room (zone AHR), battery rooms A or B (zones BR1A or BR1B), and the intermediate building-north (zone IBN-1). In each case the CCDP following equipment loss was found to be bounded by the loss of a single electrical division. Again, the suppression successes comprise all but a small fraction of fires in these compartments.

(B) Barrier and separation considerations for assuming limited damage: Five zones were analyzed specifically for limitations on the assumed damage. The results are discussed below. The analysis typically considered the combustible materials available, fire propagation paths, barriers and distances between fires and damage targets. In discussing these compartments, conditional probabilities were typically offered (conditional on the compartment fire), but without discussing the bases. One example was provided that indicated that the floor area occupied by the target was considered in making this determination.

Auxiliary building operating level (zone ABO): Primary targets are the two component cooling water (CCW) pumps separated by nine feet. Limited combustibles (grease in sealed gear boxes), particularly intervening combustibles, limit most fire damage to a single pump. Only a properly located transient combustible fire would put both pumps at risk and a conditional probability of 0.01 was assigned to this scenario. A single pump loss comprised the rest (0.99) of the damage frequency.

The plant model was rerun assuming all fires result in the loss of both pumps. The fire CDF increased by 8.8E-7 per year. This was attributed to the fact that the plant can operate on AFW alone "for a long period of time." While CCW would be needed to support RHR, in the near term considered, no PORV LOCA was expected that would create a demand for RHR.

Turbine building basement (zone TB-1): Several fire sources are located in the area. The offsite power connection to the 480 V safeguards buses are the primary targets.

- The turbine lube oil reservoir is a primary fire source, but was screened based on the presence of suppression, an oil spill/leak collection system, and large distances to targets.

- Other fire sources are the two condensate booster pumps and four air compressors. Floor area appeared to be the primary consideration in assigning a conditional probability of damage:

    - 7% of compartment frequency: damage to buses 17 and 18,
    - 3% of compartment frequency: damage to all four buses, 14, 16, 17, and 18.

Reactor containment mezzanine (zone RC-2): All fires were assumed to damage all equipment, with the exception that reactor coolant system was assumed to remain functional. A large separation was credited with precluding the loss of both reactor coolant pumps. Natural circulation is also available.

Auxiliary building basement level (zone ABB): This zone was divided into three sub-zones, one designated as the safety injection pump area and containing most of the cable targets. Propagation to this sub-zone from the other two was not considered credible.

Screen house operating level (zone SH-2): The 480 V safeguards Buses 17 and 18 and four service water (SW) pumps and their associated electrical cabinets are the primary targets. In addition to the targets, fire sources include transients and a diesel fire pump. The transients and diesel pump were assumed to have the capability to damage two SW pumps. A fire in a SW pump was assumed not to be able to damage an adjacent pump. Only a "significant" cabinet fire was assumed to be able to damage both buses, with a resulting loss of all four SW pumps. Lesser cabinet fires were assumed to impact two SW pumps. Separation and limited means of propagation were the primary considerations noted.

Conditional damage probabilities were not given for this zone.

The response noted that a sensitivity study was performed to determine the importance of the two- versus four-pump assumption for this zone. A probability of 0.001 was assigned to the loss of all four SW pumps, and the fire CDF increased by 1.5E-7 per year. (The initial value was not given.) The response was presented as supporting the observation that Ginna can shutdown without service water, for example, by using fire hoses to cool diesel generators.

The licensee response to the RAI has greatly clarified ambiguities regarding the analysis performed, but has not provided information supporting the selection of conditional probabilities of damage. The partitioning of compartment frequency into damage must be regarded as an inferior approach to explicit fire modeling, especially in areas dominated by fixed ignition sources, combustibles, and targets. This is regarded as a weakness of the study and submittal.

2.1.7 Detailed Fire PRA and Uncertainty Analysis

The detailed quantitative assessment of the remaining 19 unscreened fire zones involved identifying and quantifying potential fire scenarios in each zone. The licensee indicated that fire scenarios were developed based on the physical configuration of the fire zone, fire suppression features, and the locations of ignition sources, combustibles, equipment, and cables.

The evaluation of each fire zone required a CCDP for each identified scenario. This was accomplished using the PRA model with the equipment assumed damaged for each scenario set to failed. The CCDPs that were reported in the submittal range from 5.2E-7 to 4.0E-1. The CCDP of 4.0E-1 was reported for fires that destroy all equipment in the auxiliary building mezzanine. Such a fire causes a loss of CCW that induces an RCP seal LOCA and a loss of safety injection. The results of the detailed quantitative assessment are presented in the submittal in tabular form. A discussion of the dominant scenarios was also provided.

In the submittal, the licensee states that operator recovery actions were applied to account for local operation of components and systems due to the inability to control them from the control room. These actions include locally starting motor-driven AFW

pumps and diesel generators, locally aligning the standby AFW system, locally aligning the turbine-driven AFW pump, and locally aligning the technical support center diesel generator to provide power to the charging pumps. The HEPs for these actions were assigned without any detailed human reliability analysis (HRA) evaluation. The HEPs ranged from 0.1 to 1E-4. The HEPs for three actions were taken from the San Onofre IPEEE (the licensee states they are applicable to Ginna).

In the submittal, some of the HEPs for the human actions included in the PRA model were also adjusted from the screening values used in the quantitative screening assessment. In some cases, the HEP was set to 1.0 and the action represented in the HEP combined with the local recovery action discussed above. In other cases, the internal events PRA value for the HEP was argued to be valid for the fire scenarios. In some cases, the value of HEPs used in the fire PRA evaluation was increased above the screening values used in the quantitative screening. This appears to have been done when the screening value represented a control room action but the detailed fire PRA was modeling it as a local action. Again, when HEPs were changed, they were assigned and not evaluated using an HRA methodology. In addition, many of the HEPs from the screening assessment which were also used in the detailed fire PRA evaluation, were also assumed without an HRA evaluation.

For control room fires resulting in evacuation, Ginna procedures result in the plant being placed in a self-induced station blackout (SISBO). A description of control room evacuation scenarios provided in the submittal indicates that this SISBO condition was modeled. Specifically, the CCDP was calculated to include failure to open the offsite power breakers to the emergency buses and manually starting and loading the diesel generators.

In the original analysis, the core damage frequencies for the remaining 19 fire zones resulting from this analysis range from 6.4E-8/yr to 1.2E-5/yr. The total core damage frequency for these fire zones is 6.4E-5/yr. (This estimate was updated, as described below.) Fires leading to a LOCA and loss of safety injection contributed 25% of this CDF while fires leading to a transient with loss of core cooling comprise 23% of the CDF. No uncertainty analysis was performed in the fire assessment. However, the submittal provides a qualitative discussion of the sources of uncertainty.

The fire PRA evaluation of the unscreened fire zones appears to have utilized estimates of the CCDPs based on assumed damage levels and estimates of HEPs. The submittal also noted one plant improvement and a total of five modifications that were being considered. However, it was not clear whether these changes had been credited in the study, or which changes had been implemented. The status of the proposed plant modifications credited in the fire study was the subject of an RAI to the licensee. In its response to this RAI (RAI #8), the licensee described a detailed update of the fire risk estimate. This update was based on an improved evaluation that included explicitly modeled operator actions and suppression.

The relevance of the summary of the updated evaluation to RAI #8 is that it was the basis for dropping the five modifications described in the submittal. Only the single plant improvement was made, although in addition to this credit, the modifications were credited in the original submittal. The single improvement was implemented in March 1999 with the installation of fuses in control circuits routed through the screen house.

The fire risk estimate following the revision is 3.3E-5 per year, about half of the original estimate. In addition, risk importance (both Fussell -Vesely and Risk Achievement Worth) was determined for initiating events, operator errors, testing and maintenance outages, systems, components, and modeling assumptions. Sensitivity analyses are described below. The contributions to risk are tabulated below by major plant area.

Plant Area	**% Fire CDF**
Control Building	42.2
Turbine Building	25.5
Auxiliary Building	12.2
Transformer Yard	5.8
Diesel Generator Rooms	4.6
Containment	4.3
Screenhouse	3.1
Intermediate Building	1.2
Cable Tunnel	0.8
Tech Support Center	0.1

2.1.8 Sensitivity and Importance Ranking Studies

In addition to the sensitivity studies discussed above, the licensee performed five sensitivity studies, three of which pertain to the electrical cabinet fires. As previously discussed, the licensee assumed a probability of 0.1 that fires initiated in cabinets would propagate outside the cabinet and damage surrounding equipment. In one sensitivity analysis, the licensee increased this probability to 0.5 which increased the overall CDF by a factor of 2.2. In the second sensitivity analysis, this probability was reduced to 0.01. The licensee indicates the fire-related CDF decreased by 21%. In the third sensitivity study, cabinet fires were assumed not to destroy everything in the cabinet. This was modeled by reducing the CCDP for fires that don't progress beyond the cabinet

by a factor of 0.1. The overall CDF reported in the submittal for this case was reduced by a factor of 9%.

A fourth sensitivity study was performed to assess the impact that manual suppression has on the fire-related core damage results. The licensee reported that, when credit was taken for manual suppression only in the control room, the CDF increased by a factor of two. In the fifth sensitivity study, the risk reduction for proposed plant modifications to reduce spurious opening of three valves was determined. Spurious opening of MOV 857B was modeled as failing RHR aligned for shutdown cooling. Spurious opening of MOVs 850A and 850B was modeled as causing draining of the RWST volume into the containment sump (significant draining can cause cavitation of the safety injection pumps if the pump suction source is not transferred). If a modification is undertaken to reduce the spurious opening of these valves, the licensee states that the fire-related CDF would decrease by 25%.

The licensee response to RAI #8 was previously discussed and included sensitivity studies for operator errors (which were modeled explicitly in the updated study), hot-short-induced failures, and truncation limits. The results were given for the assumptions of all failures true (failed action, or failed equipment) and all failures false with the following results:

	True	False
Operator error	CDF = 0.494/yr	CDF down 69%
Hot-short failures	CDF up 61%	CDF down 6%

Truncation sensitivity showed 93% of the fire CDF was comprised of cutsets contributing greater than 1E-9/year to the risk. The conclusion drawn from the sensitivity studies was that operator error is important at Ginna.

2.2 Special Issues

As a part of the IPEEE fire submittal, the utilities were asked to address a number of fire-related issues identified in the Fire Risk Scoping Study (FRSS) [12], USNRC Unresolved Safety Issues (USI), and Generic Safety Issues (GSI). Specific review guidance on these issues is taken from Reference 6.

2.2.1 Decay Heat Removal (USI A-45)

As discussed in Generic Letter 88-20 [4] and NUREG 1407 [5], USI A-45, which is associated with the adequacy of decay heat removal at nuclear power plants, is subsumed into the IPE submittals. A submittal meeting the intent of Generic Letter 88-20, Supplement 4 is assumed to satisfy the requirements of USI A-45. Specifically, the fire assessment presented in the IPEEE submittal should address the adequacy of long-term decay heat removal in the event of fires.

The Ginna submittal addresses this issue by first identifying the systems available for removing decay heat at Ginna. Decay heat removal can be accomplished through the steam generators using either the main feedwater system, the AFW system, or the standby AFW system. The RHR system is used for closed cycle cooling (i.e., shutdown cooling) or for sump recirculation. The licensee further addresses this issue by stating that all fire-related core damage sequences involve challenge of decay heat removal systems and the potential failure of decay heat removal due to both fire impacts and random failures. The submittal indicates that 47% of the fire-related CDF involves failure of decay heat removal. The licensee concludes that no new vulnerabilities were identified from the fire assessment and thus USI A-45 is judged to be resolved.

2.2.2 Effects of Fire Protection System Actuation on Safety-related Equipment (FRSS, GSI 57, MSRP)

This issue is associated with the concern that traditional fire PRA methods have generally considered only direct thermal damage effects. Other potential damage mechanisms have not been addressed, such as smoke and the potential that the activation of fire suppression systems, either as part of actual fire fighting or spuriously, might result in damage to plant systems and components. In general, this is an area where the database on equipment vulnerability is rather sparse. The analytical results obtained for resolution of the issue, subsumed by GSI 57, identified the dominant risk contributors as: (1) seismic-induced fire plus seismic-induced suppressant diversion and (2) seismic-induced actuation of the fire protection system (FPS). The NRC anticipated that the licensee would conduct seismic/fire walkdowns to assess (1) whether an actuated FPS would spray safety-related equipment, and (2) whether some protective measures to prevent the same could be instituted. The results could be documented in the IPEEE submittal.

The submittal indicates that the effect of inadvertent actuation of fire suppression systems on safety-related equipment is addressed by the Cause Investigation and

Corrective Action Plan, CAR-1869, “Fire Suppression Systems Effects.” The licensee indicates that this report made four recommendations which were later implemented. They were not listed in the submittal.

2.2.3 Fire-induced Alternate Shutdown/Control Room Panel Interactions (FRSS, GSI 147)

The issue of control systems interactions is associated primarily with the potential that a fire in the plant, e.g., the main control room (MCR), might lead to potential control systems vulnerabilities. Given a fire in the plant, the likely sources of control systems interactions are between the control room, remote shutdown panel, and shutdown systems. Specific areas that should be addressed in the IPEEE fire analysis include 1) Electrical independence of the remote shutdown control systems; 2) Loss of control equipment or power before transfer; 3) Spurious actuation of components leading to component damage, LOCA, or interfacing LOCA; and 4) Total loss of system function. It is anticipated that the licensee’s submittal will describe its remote shutdown capability including the nature and location of the shutdown station(s) and the types of control actions that can be taken from the remote panel(s).

The licensee indicated in the submittal that a review of the control and monitoring circuits of the plants was conducted to verify that safe shutdown circuits have been physically located independent of, or can be isolated from, the control room. An analysis of control room evacuation and remote shutdown capabilities was performed and documented in the tier-2 document. No significant fire vulnerability was identified by the licensee.

As previously discussed, the submittal indicates that hot shorts were modeled in the assessment. The discussion above of the reevaluation of the plant fire risk in response to RAI #8 noted that hot shorts appear in the plant model as simple failures of systems and components. A broader question (RAI #6) asked the licensee to address the four aspects above of the issue of control system interactions. The response is described in the following paragraphs.

The response noted that alternative shutdown (ex-control room) is prescribed by Procedure AP-CR.1 as the result of an unsuppressed fire in any of the following areas:

- the control complex: control room, relay room, and air handling room,
- the cable tunnel,
- the auxiliary building mezzanine and basement,
- battery room A,
- battery room B.

Eight staff members are required for alternate shutdown, but only six are required during the first hour. Two additional operators are assigned to the fire brigade and are expected to be relieved after one hour, either as the result of the fire being extinguished,

or by recalled operators. Two Primary Shutdown Stations would be continuously occupied during a shutdown, six Support Stations would be staffed as needed, and several valve locations would be staffed for short periods of manual operation and status verification, as needed.

Electrical independence: The description of operations performed at each location includes transfer of control and isolation of power sources from the presumed fire-impacted circuits. Some available local indications are listed in these descriptions. Several local operations are also noted.

Loss of control before transfer, spurious operation, loss of function: The response states that the alternate shutdown plan reduces concerns regarding spurious operations and loss of control, based on the following:

- Alternative shutdown circuits are electrically isolated from fire areas of concern.
- Power is removed from unnecessary AC and DC loads at buses and distribution panels.
- Instrument air is isolated to prevent spurious valve operation.
- Alternate shutdown utilizes separate, isolated power sources.
- Alternate shutdown circuits are separated from the fire areas of concern by rated barriers and seals.

Several operations at remote stations described in the response noted the intention of precluding spurious actuations. The operators' actions in performing a remote shutdown were specifically modeled in quantifying the fire risk using the updated plant model. Hence, HEPs associated with a remote shutdown are included in the results described previously.

The licensee has presented an appropriate response to this question.

2.2.4 Smoke Control and Manual Fire Fighting Effectiveness (FRSS, GSI 148)

The issue of smoke control and manual fire fighting effectiveness is associated with the concern that nuclear power plant ventilation systems are known to be poorly configured for smoke removal in the event of a fire, and hence, a significant potential exists for the buildup of smoke to hamper the efforts of the manual fire brigade to suppress fires promptly and effectively. Sensitivity studies have shown that prolonged fire fighting times can lead to a noticeable increase in fire risk. Smoke, identified as a major contributor to prolonged response times, can also cause misdirected suppression efforts and hamper the operator's ability to shut down the plant safely.

As indicated in this document, manual fire suppression was credited in the Ginna fire assessment. The submittal indicates that a review was conducted of the Ginna fire fighting programs, procedures, and training records to confirm and document that the plant has an effective fire fighting team. The procedures establishing fire brigade training and drill requirements, evacuation plans, and inspection requirements are

identified in the submittal. The submittal also provides a table of fire drill records that shows that the average time from the time of an alarm until the drill was completed is 13 minutes. Based upon this review, the licensee concludes that the manual fire fighting capability at Ginna is adequate.

With regard to operator effectiveness in performing manual shutdown actions, the licensee evaluated the normally operating plant ventilation systems and their effect on the ability of the operators to access and operate equipment needed to shutdown the plant. The results of this evaluation are not provided in the submittal.

2.2.5 Seismic/Fire Interactions (FRSS, MSRP)

The issue of Seismic/Fire Interactions primarily involves three concerns. First is the potential that seismic events might result in fires internal to the plant. Such threats might be realized from inadequately secured liquid fuel or oil tanks, through breakage of fuel lines, or through the rocking of unanchored electrical panels (either safety or non-safety grade). The second concern is the potential that seismic events might render fixed fire suppression systems inoperable. This could include detection systems, fixed suppression systems, and fixed manual fire fighting support elements such as the plant fire water distribution system. The third concern is that a seismic event might spuriously actuate fixed fire detection and suppression systems. The spurious operation of detectors might both complicate operator response to the seismic event and/or cause the actuation of automatic fire suppression systems. Actuation of a suppression system may lead to flooding problems, habitability concerns (in the case of CO_2 systems), the diversion of suppressants to non-fire areas rendering them unavailable in the event of a fire elsewhere, the potential over-dumping of gaseous suppressants resulting in an overpressure of a compartment, and spraying of important plant components. It had been anticipated that a typical fire IPEEE submittal would provide for some treatment of these issues through a focused seismic/fire interaction walkdown.

The Ginna submittal states that the potential for seismically-induced fires, seismic actuation of fire suppression systems, and seismic degradation of fire suppression systems was addressed in the seismic IPEEE. Plant walkdowns were performed to address these issues. Five potential vulnerabilities were identified and resolved. First, a house heating boiler was found to be inadequately anchored and thus could shift during an earthquake causing damage to the attached natural gas line. The licensee indicates a design change will be implemented to anchor the boiler (no implementation date was given in the submittal). Second, several locations were identified where block wall failures could rupture a hydrogen line or topple hydrogen cylinders. The licensee states that the hydrogen line is not a fire risk since it is not valved on during power operation. The licensee states that the hydrogen cylinders do not pose a risk since the hydrogen is diluted with nitrogen and any release would thus result in a low hydrogen concentration.

Failure of block walls was also identified as potentially causing the actuation of two fire suppression systems. The licensee indicates that the inadvertent actuation of a deluge system in the relay room would not have a significant impact since the relay cabinets are

closed on top and the cable penetrations are sealed. Actuation of a deluge system in the intermediate building would only spray the turbine-driven AFW pump. Seismic actuation of a pre-action system in the same area was dismissed because of the existence of fusible link sprinkler heads in the system. The submittal indicates that since block walls are fire barriers, failure of walls between the service and intermediate buildings and between the turbine and intermediate buildings during an earthquake could impact the fire protection of safety-related equipment. The licensee addresses these failures by stating that the potential for fires initiated in these areas is small.

The final vulnerability addresses the fact that two reactor coolant pump oil collecting tanks in the containment were not reviewed during the seismic walkdown due to inaccessibility. The seismic capacity of the tanks was subsequently verified by the licensee through a design evaluation. The licensee states that further verification would be performed when a walkdown is performed the next time the containment is accessible.

In summary, the licensee states that the potential for seismic/fire interactions does not represent a seismic or fire vulnerability.

2.2.6 Adequacy of Fire Barriers (FRSS)

The common reliance on fire barriers to separate redundant components needed to achieve a safe shutdown has elevated the risk sensitivity of fire barrier performance. Degraded fire barrier penetration seals and unsealed penetrations in some barriers can contribute to this source of fire risk, since fires in one area might impact other adjacent or connected area through the spread of heat and smoke. In general, it is expected that a utility analysis would provide for some treatment of such potential by considering that (1) manual fire fighting activities might allow for the spread of smoke and heat through the opening of access doors, and (2) that the failure of active fire barrier elements such as normally open doors, water curtains, and ventilation dampers might compromise barrier integrity.

The submittal indicates operability and maintenance of fire dampers is covered under existing procedures. Ten percent of the dampers are checked every year. The licensee indicates that to date, there has been no failure of fire doors, fire dampers, or penetration seals that have not been promptly detected during daily plant tours. Thus, based on the review of test results and inspection, testing and maintenance technical requirements, the licensee concludes that the fire barriers at Ginna are effective and adequate.

It was noted earlier that the Ginna credited fire wrap in the fire analysis. This fire wrap is apparently not Thermo-Lag. The licensee notes that the fire wrap installed on cables associated with charging pump A is a plant feature that provides a significant reduction in the fire risk. It was also noted earlier that no criterion was established in the submittal for determining the potential for propagation between fire zones.

2.2.7 Effects of Hydrogen Line Ruptures (MSRP)

The use of flammable gases in the plant, including hydrogen, introduces the potential that a rupture of the gas flow lines might lead to the introduction of a serious fire hazard into plant safety areas. It had been anticipated that a typical fire IPEEE analysis would include the consideration of such sources in the analysis.

As indicated in Section 2.2.5 of this review, the Ginna submittal discusses seismic-initiated fires involving flammable gases. In addition, in a separate letter attached to the submittal, a potential explosion in a hydrogen storage area is addressed. The concern is the proximity of the hydrogen storage area to storage areas for turbine lube oil and emergency diesel fuel oil. Based on the existence of reinforced walls between these areas and separation distances, this issue was resolved. In addition, a licensee evaluation of the core damage potential due to an explosion of the hydrogen bottles in this area resulted in a low CDF.

2.2.8 Common Cause Failures Related to Human Errors (MSRP)

Common cause failures resulting from human errors include operator acts of omission or commission that could be initiating events or could affect redundant safety-related trains needed to mitigate other initiating events. It had been anticipated that a typical fire IPEEE analysis would include the consideration of such failures in the submittal.

This issue was not explicitly discussed. As indicated in Section 2.1.7 of this review, operator recovery actions were credited in the fire PRA portion of the detailed assessment. Some of the HEPs for the operator actions modeled in the internal event PRA used in the fire assessment were modified by the licensee to account for fire impacts. However, no detailed HRA was performed to calculate the HEPs; the HEPs were assumed. The licensee indicates that dependencies between these HEPs were accounted for in the study.

2.2.9 Non-safety Related Control System/Safety Related Protection System Dependencies (MSRP)

Multiple failures in non-safety-related control systems may have an adverse impact on safety-related protection systems as a result of potential unrecognized dependencies between control and protection systems. The licensee’s IPE process should provide a framework for systematic evaluation of interdependence between safety-related and non-safety related systems and identify potential sources of vulnerabilities. It had been anticipated that the fire IPEEE analysis would include the consideration of such dependencies in the submittal.

This issue was not explicitly discussed. The fire assessment utilized the Ginna IPE model which includes both safety-related and non-safety-related systems and the dependencies between systems. In addition, in the evaluation of HEPs included in the fire evaluation, the impact of fire-induced failure of indication was included.

2.2.10 Effects of Flooding and/or Moisture Intrusion on Non-Safety- and Safety-Related Equipment (MSRP)

Flooding and water intrusion events can affect safety-related equipment either directly or indirectly through flooding or moisture intrusion of multiple trains of non-safety-related equipment. This type of event can result from external flooding events, tank and pipe ruptures, actuations of the fire suppression system, or backflow through part of the plant drainage system. It had been anticipated that the fire IPEEE analysis would include the consideration of such events in the submittal.

This issue was not explicitly discussed. However, the impact of inadvertent fire suppression system actuation is discussed in Sections 2.2.2 and 2.2.5 of this report.

2.2.11 Shutdown Systems and Electrical Instrumentation and Control Features (SEP)

The issue of shutdown systems addresses the capacity of plants to ensure reliable shutdown using safety-grade equipment. The issue of electrical instrumentation and control addresses the functional capabilities of electrical instrumentation and control features of systems required for a safe shutdown, including support systems. These systems should be designed, fabricated, installed, and tested to quality standards and remain functional following external events. It had been anticipated that the fire IPEEE analysis would include the consideration of this issue in the submittal.

Section 2.2.3 of this review discusses the submittal and RAI responses on the issue of control system interactions.

2.3 Containment Performance Issues Unique to Fire Scenarios

The licensee addresses the containment performance during a fire by reviewing the potential for containment bypass, determining the potential impacts on containment isolation, and by reviewing the availability of containment heat removal systems during fire events. The licensee states there are no new containment failure modes introduced by fire events.

The licensee addresses the potential for containment bypass due to both interfacing system LOCAs (ISLOCAs) and induced steam generator tube ruptures (SGTR). Two ISLOCA scenarios are identified. The first, involving two AOVs in the letdown line, is dismissed based on the fact that (1) hot shorts of the control cables for both AOVs would be required and (2) this event can be mitigated by closing a manual valve downstream or by isolating the instrument air line to the valves. The second scenario involves two MOVs which isolate the RHR from the RCS. Since circuit breakers for these valves are locked-open, an ISLOCA can only be initiated by a three-phase hot short which the licensee states is unlikely. An induced SGTR is judged by the licensee to be unlikely since it would require failure of a tube and a failed-open main steam isolation valve.

The licensee addresses containment isolation by noting that isolation valves can fail open during a fire due to hot shorts but usually can be closed by either manually closing them (both AOVs and MOVs) or by removing the air supply (AOVs). The licensee also states that a fire could damage the containment isolation actuation system but due to the time available to actuate the system, manual containment isolation can be performed.

The submittal indicates that the proportion of the fire-related accident sequences for which containment heat removal fails is 47% which corresponds to a CDF of 3.0E-5/yr. The licensee judges this to be acceptable due to the conservative nature of the fire analysis and the potential for recovery due to the significant time available during these accidents.

2.4 Plant Vulnerabilities and Improvements

The licensee indicated in the submittal that the results of fire PRA were compared against the Nuclear Energy Institute (NEI) Severe Accident Closure Guidelines to determine if vulnerabilities exist, thus implicitly defining the term “fire vulnerability.” Based on the NEI criteria, the licensee concluded that the fire assessment confirms that there are no fire-induced vulnerabilities. However, several plant and procedural modifications were identified as a result of the analysis. One modification was planned for implementation and was credited in the analysis (An RAI response indicated implementation of this modification in March 1999.):

- Fuses will be installed on control circuits routed in the screen house associated with the functioning of 4160 VAC circuit breakers. The fuses will be designed to open if grounding occurs during a fire thus permitting the protective function of the circuit breakers to remain intact.

Several modifications were considered by the licensee at the time of the submittal, but were dismissed based on the requantification of the plant fire risk described in the response to RAI#8:

- An operating procedure enhancement was proposed by the licensee for performing local recovery of the pressurizer heaters if control of the heaters is lost from the control room. (The pressurizer heaters are one means of providing long-term RCS circulation.)

- A spurious opening of MOV 857B fails RHR shutdown cooling. The insertion of a warning in the alternate shutdown procedure ER-FIRE-1 was being considered by the licensee to indicate that this valve can be closed locally.

- Installation of additional sealed containers for transient combustibles storage in the auxiliary building basement was being considered by the licensee.

- Spurious opening of MOVs 850A and 850B due to hot shorts can lead to draining of the RWST volume into the containment sump. Methods to reduce this potential were being considered by the licensee.

- Installation of a local pressure gauge to permit RWST level measurement in the event of fire-induced damage to level instrumentation was being considered by the licensee. Such a modification would assist the operator in switching to sump recirculation.

3.0 CONCLUSIONS AND RECOMMENDATIONS

The Ginna fire assessment was performed by application of the FIVE and PRA methodologies. The licensee stated that the qualitative and quantitative screening phases of the assessment conformed to the FIVE methodology. The assessment also discusses the FRSS issues and USI A-45. Some strengths of the analysis include:

- Plant walkdowns of quantitatively screened areas to confirm equipment locations were performed.
- Plant-specific data was used to update generic fire frequencies based on a two-stage Bayesian methodology.
- The analysis utilizes the internal events PRA models which include both Appendix R and non-Appendix R equipment. The location of this equipment (including the associated cabling) was apparently determined and used in the definition of fire scenarios.
- The HEPs for human errors included in the internal events PRA model were reviewed for impacts from the fire, including impacts on alternate shutdown. In response to an RAI, a fire-specific human reliability analysis was performed.
- The impact of hot short failures was included in the fire assessment.
- Several sensitivity studies were performed to evaluate the importance of key parameters.

The review of the original submittal raised a number of questions. Subject areas found deficient during the review of the submittal led to requests for additional information. The licensee response resolved the outstanding issues. Residual weaknesses in the Ginna submittal whose effects on the results are minimal, or understood and do not require further elaboration include the following:

- No fire modeling was described. It is not clear how conditional damage probabilities were determined for fires in a given compartment. (RAI #3)
- Optimistic reliability estimates may have been used for the installed fire suppression systems. (RAI #7)

Overall the submittal is considered to be an adequate response to the IPEEE requirements. Based on the documentation provided in the Ginna IPEEE submittal and responses to RAIs, it appears that a sufficient level of documentation and appropriate bases for analysis have been established to conclude that the subject licensee submittal has substantially met the intent of the IPEEE process.

4.0 REFERENCES

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Attachment 3

R.E. GINNA NUCLEAR POWER PLANT

INDIVIDUAL PLANT EXAMINATION OF EXTERNAL EVENTS (IPEEE)

TECHNICAL EVALUATION REPORT

HIGH WINDS, FLOODS, AND OTHER EXTERNAL EVENTS

Brad Hardin, USNRC
October 4, 2000

R.E. GINNA NUCLEAR POWER PLANT
INDIVIDUAL PLANT EXAMINATION OF EXTERNAL EVENTS (IPEEE)
TECHNICAL EVALUATION REPORT
HIGH WINDS, FLOODS AND OTHER (HFO) EXTERNAL EVENTS

1.0 Introduction

Ginna is a single-unit, Westinghouse two-loop PWR located approximately 20 miles northeast of Rochester, New York on the south shore of Lake Ontario. The plant is owned by Rochester Gas and Electric Company (RG&E) and began commercial operation in July 1970. The plant has a power rating of 470 MWe.

Since Ginna was designed prior to the 1975 SRP criteria, technically it is not a 1975 SRP plant. However, the licensee's HFO IPEEE process used the progressive screening approach described in NUREG-1407 focussed on demonstrating that the design and construction of the plant in the HFO areas meets the 1975 SRP criteria. The licensee's evaluation included a plant walkdown specifically focused on the HFO topics to confirm that no plant changes had occurred since the issuance of the original Operating License that would impact on the HFO areas of review.

2.0 High Winds

In Reference 8.3, the licensee (RG&E) stated that, as a part of the Systematic Evaluation Program (SEP), Ginna was reviewed against the criteria of the 1975 Standard Review Plan (SRP), specifically SEP topics II-2.A, III-2, III-4.A, and III-7.B. These topics include consideration of high winds, tornadoes, tornado missiles, and structural load combinations. Additional supporting information regarding the review of high winds was included in Reference 8.4. After the SEP review, RG&E made several modifications to the plant to increase the protection from high winds. These modifications are described in Section 3.3 of the staff's R.E. Ginna safety evaluation report (FSAR) and included the reinforcement of several block walls and the addition of reinforced structural members in the turbine building. Also, in a safety evaluation report dated August 22, 1983, the NRC indicated that the licensee's approach for increasing the protection against high winds was acceptable. In Reference 8.3, the licensee states that a site walkdown was performed coupled with a rereview of the NRC's 1983 SER results, and no changes at the plant were found that would increase the plant's vulnerability to high winds. In their submittal, the licensee stated that the plant is in compliance with the criteria given in Section 5.2.4 of NUREG-1407.

3.0 External Floods

In Reference 8.5, the licensee submitted its external flooding analysis including the assessment of the revised levels for probable maximum precipitation. The licensee reported that in evaluating the new Probable Maximum Precipitation (PMP) criteria it was determined that on-site flooding and roof ponding due to the increased rainfall did not pose a significant threat to the plant with the exception that one building had a marginal roof design. This building was the Control Building for which the building parapet was to be modified or an additional scupper added to minimize ponding on the roof. Plant walkdowns were performed to (1) verify the adequacy of existing flood protection measures, and (2) evaluate the site and local plant area to assess the impact of the new rainfall levels. It was determined that a site area change involving

a security vehicle barrier system has increased the plant's resistance to flooding, and that no other significant changes were identified. During the walkdown, the effects of flooding of Deer Creek that flows into Lake Ontario was reviewed and it was concluded that the existing plant protection measures for flooding were adequate considering the PMP.

4.0 **Transportation and Nearby Facility Accidents**

In Reference 8.3, the licensee states that this topic was specifically addressed within SEP Topic II-1.c, "Potential Hazards due to Nearby Transportation, Industrial and Military Facilities." It is also stated that issues related to this topic were reviewed against the criteria of Sections 2.2.1 and 2.2.2 of the 1975 SRP, and it was determined that Ginna Station met these criteria. In Reference 8.3, the licensee states that a site walkdown was performed with a rereview of the SEP criteria relating to this issue, and no changes at the plant were found that would increase the plant's vulnerability to transportation and nearby facility accidents. In Reference 8.4, the licensee submitted additional supporting information regarding the transportation and nearby facility accidents.

5.0 **Other External Events**

The licensee considered the need to perform additional evaluations on the potential effects of other external events including lightning, ice, hail, snow, and dust, and no additional areas were identified where the plant was vulnerable to these effects. The licensee reported that in an earlier plant modification, the power for the heaters on the cooling water intake screens on Lake Ontario had been increased to protect against ice formation (slush). The change in screen heater power is reported in Inspection Report 0502442000-1.

6.0 **Generic Safety Issue (GSI) Resolution**

GSI-103, "Design for Probable Maximum Precipitation"

In Reference 8.5, the licensee reported its evaluation of the new Probable Maximum Precipitation (PMP) criteria where it was determined that on-site flooding and roof ponding due to the increased rainfall did not pose a significant threat to the plant with the exception that one building had a marginal roof design. This building was the Control Building for which the building parapet was to be modified or an additional scupper added to minimize ponding on the roof. The licensee used the National Weather Service's revised Hydrometeorological Report No. 52 to determine that the value of PMP for the Ginna site is 16.5 inches over one-square mile in one hour. The staff finds that the licensee's GSI-103 evaluation is consistent with the guidance provided in Section 6.2.2.3 of NUREG-1407 and, therefore, this issue is resolved.

GSI-156, "Systematic Evaluation Program" (SEP)

The licensee's IPEEE submittal contains information to directly address the following external-event-related SEP issues: (1) dam integrity and site flooding (Section 5.2 of Reference 8.1 and Reference 8.5); (2) site hydrology and ability to withstand floods (Reference 8.5); (3) industrial hazards (References 8.3 & 8.4); (4) tornado missiles (Reference 8.3); (5) severe weather effects on structures (Reference 8.5); and (6) design codes, criteria and load combinations (Reference 8.4). Based on the results of the IPEEE submittal review, the staff considers that the licensee's process is reasonable and is capable of identifying potential vulnerabilities associated with this

issue. On the basis that no vulnerabilities associated with this issue were identified in the licensee's IPEEE submittal, the staff considers this issue resolved for Ginna.

GSI-172, "Multiple System Responses Program (MSRP)"

With respect to the MSRP issue related to the HFO area which is entitled "effects of flooding and/or moisture intrusion on non safety-related and safety-related equipment," the licensee addressed this issue in Section 5.6.2 of its Reference 8.1 submittal, and in Section 3.9.3 of its Reference 8.2 submittal. The licensee reported that their evaluation did not identify any potential vulnerabilities associated with this issue. Based on the overall results of the staff's IPEEE submittal review, the staff considers that the licensee's process is capable of identifying potential vulnerabilities associated with this issue. On the basis that no potential vulnerability regarding this issue was identified in the licensee's submittal, the staff considers the HFO-related aspects of this issue to be resolved for Ginna.

7.0 **Conclusions**

A strength in the IPEEE submittal is that Ginna is an SEP plant, and was subjected to a detailed review for SEP much of which is applicable to IPEEE.

It is concluded that the licensee's submittal for the Ginna Station regarding the part of the IPEEE HFO review relating to high winds, floods, transportation and other events meets the intent of GL 88-20.

8.0 **References**

8.1 IPEEE Seismic Evaluation Report, Attachment to letter from Robert C. Mecredy, Rochester Gas and Electric Corporation, to USNRC, January 31, 1997.

8.2 IPEEE Fire Analysis Report, Attachment to letter from Robert C. Mecredy, Rochester Gas and Electric Corporation, to USNRC, June 30, 1998.

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8.5 IPEEE Probable Maximum Precipitation Analysis Report, Attachment to letter from Robert C. Mecredy, Rochester Gas and Electric Corporation, to USNRC, December 21, 1998.

8.6 Response to February 23, 1996 Request for Additional Information (RAI) (Internal Flooding Issues Only), Attachment to letter from Robert C. Mecredy, Rochester Gas and Electric Corporation, to USNRC, March 1, 1999.

8.7 Request for Additional Information on the R.E. Ginna Nuclear Power Plant IPEEE Submittal (TAC No. M83624), Letter from Guy S. Vissing, USNRC, to Robert C. Mecredy, Rochester Gas and Electric Corporation, February 25, 1999.

8.8 Response to Request for Additional Information on IPEEE, Attachment to letter from Robert C. Mecredy, Rochester Gas and Electric Corporation, to USNRC, July 30, 1999.

Attachment 4

R.E. GINNA NUCLEAR POWER PLANT

INDIVIDUAL PLANT EXAMINATION OF EXTERNAL EVENTS (IPEEE)

STAFF EVALUATION REPORT

INTERNAL FLOODING

Erasmia Lois, USNRC
November 7, 2000

R.E. GINNA NUCLEAR POWER PLANT
INDIVIDUAL PLANT EXAMINATION OF EXTERNAL EVENTS (IPEEE)
TECHNICAL EVALUATION REPORT
HIGH WINDS, FLOODS AND OTHER (HFO) EXTERNAL EVENTS

1.0 Introduction

Ginna is a single-unit, Westinghouse two-loop PWR located approximately 20 miles northeast of Rochester, New York on the south shore of Lake Ontario. The plant is owned by Rochester Gas and Electric Company (RG&E) and began commercial operation in July 1970. The plant has a power rating of 470 MWe.

Since Ginna was designed prior to the 1975 SRP criteria, technically it is not a 1975 SRP plant. However, the licensee's HFO IPEEE process used the progressive screening approach described in NUREG-1407 focussed on demonstrating that the design and construction of the plant in the HFO areas meets the 1975 SRP criteria. The licensee's evaluation included a plant walkdown specifically focused on the HFO topics to confirm that no plant changes had occurred since the issuance of the original Operating License that would impact on the HFO areas of review.

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In Reference 8.5, the licensee submitted its external flooding analysis including the assessment of the revised levels for probable maximum precipitation. The licensee reported that in evaluating the new Probable Maximum Precipitation (PMP) criteria it was determined that on-site flooding and roof ponding due to the increased rainfall did not pose a significant threat to the plant with the exception that one building had a marginal roof design. This building was the Control Building for which the building parapet was to be modified or an additional scupper

added to minimize ponding on the roof. Plant walkdowns were performed to (1) verify the adequacy of existing flood protection measures, and (2) evaluate the site and local plant area to assess the impact of the new rainfall levels. It was determined that a site area change involving a security vehicle barrier system has increased the plant's resistance to flooding, and that no other significant changes were identified. During the walkdown, the effects of flooding of Deer Creek that flows into Lake Ontario was reviewed and it was concluded that the existing plant protection measures for flooding were adequate considering the PMP.

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The licensee's IPEEE submittal contains information to directly address the following external-event-related SEP issues: (1) dam integrity and site flooding (Section 5.2 of Reference 8.1 and Reference 8.5); (2) site hydrology and ability to withstand floods (Reference 8.5); (3) industrial hazards (References 8.3 & 8.4); (4) tornado missiles (Reference 8.3); (5) severe weather effects

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