September 22, 2005

Mrs. Mary G. Korsnick
Vice President R.E. Ginna Nuclear Power Plant
R.E. Ginna Nuclear Power Plant, LLC
1503 Lake Road
Ontario, NY 14519

SUBJECT: R.E. GINNA NUCLEAR POWER PLANT - AMENDMENT RE: APPLICATION OF LEAK-BEFORE-BREAK METHODOLOGY FOR PRESSURIZER SURGE LINE AND ACCUMULATOR LINES (TAC NO. MC4929)

Dear Mrs. Korsnick:

The Commission has issued the enclosed Amendment No. 92 to Renewed Facility Operating License No. DPR-18 for the R.E. Ginna Nuclear Power Plant. The amendment consists of changes to the Updated Final Safety Analysis Report (UFSAR) in response to your application transmitted by letters dated September 30, 2004, and May 28, 2005.

The amendment revises information in the UFSAR regarding the application of "leak-before-break" methodology for the emergency core cooling system accumulator lines A and B and the pressurizer surge line. The amendment permits the exclusion of these lines from the evaluation of the dynamic effects associated with postulated high-energy line breaks in the analyzed segments of the accumulator lines piping system and the pressurizer surge line piping system.

A copy of the related Safety Evaluation is also enclosed. A Notice of Issuance will be included in the Commission's biweekly *Federal Register* notice.

Sincerely,

/RA/

Patrick D. Milano, Sr. Project Manager, Section 1
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-244

Enclosures: 1. Amendment No. 92 to Renewed License No. DPR-18
2. Safety Evaluation

cc w/encls: See next page

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Accession Number: ML052430343

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DATE	08/31/05	09/08/05	09/08/05	07/07/05	09/20/05	09/21/05

Official Record Copy

DATED: September 22, 2005

AMENDMENT NO. 92 TO RENEWED FACILITY OPERATING LICENSE NO. DPR-18
R.E. GINNA NUCLEAR POWER PLANT

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R.E. GINNA NUCLEAR POWER PLANT, LLC

DOCKET NO. 50-244

R.E. GINNA NUCLEAR POWER PLANT

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 92
Renewed License No. DPR-18

1. The Nuclear Regulatory Commission (the Commission or the NRC) has found that:

   A. The application for amendment filed by the R.E. Ginna Nuclear Power Plant, LLC (the licensee) dated September 30, 2004, and May 28, 2005, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance: (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, by Amendment No. 92, the license is amended by changes to the Updated Final Safety Analysis Report (UFSAR). The licensee is authorized to change the design for emergency core cooling system accumulator lines and the pressurizer surge line regarding the exclusion of consideration of the dynamic effects associated with the postulated rupture of the analyzed segments of the accumulator piping and the pressurizer surge line piping from the current licensing basis. These changes reflect the NRC staff's approval of the analyses in Westinghouse Report WCAP-16311-P, Revision 0, "Technical Justification for Eliminating Pressurizer Surge Line Rupture as the Structural Design Basis for the R. E. Ginna Nuclear Power Plant," and Structural Integrity Associates Report SIR-99-036, Revision 0, "Leak-Before-Break Evaluation of Portions of the Accumulator A and B Piping at R. E. Ginna Nuclear Power Station."

3. This license amendment is effective as of the date of its issuance. In the next update of the UFSAR required by 10 CFR 50.71(e), the licensee will implement this amendment by incorporating into the UFSAR the revisions to the statements on the dynamic and environmental effects from the analyses of high-energy lines as described in its September 30, 2004, and May 28, 2005, letters and as evaluated in the staff's Safety Evaluation dated September 22, 2005.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Richard J. Laufer, Chief, Section 1
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Date of Issuance: September 22, 2005

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 92 TO RENEWED FACILITY

OPERATING LICENSE NO. DPR-18

R.E. GINNA NUCLEAR POWER PLANT, INC.

R.E. GINNA NUCLEAR POWER PLANT

DOCKET NO. 50-244

1.0 INTRODUCTION

By letters dated September 30, 2004, and May 28, 2005 (Agencywide Documents Access and Management System Accession Nos. ML042860138 and ML051580149), the R.E. Ginna Nuclear Power Plant, LLC (the licensee), submitted a request for changes to the R.E. Ginna Nuclear Power Plant (Ginna) Updated Final Safety Analysis Report (UFSAR). The requested changes would allow the use of the "leak-before-break" (LBB) methodology for portions of the emergency core cooling system (ECCS) accumulator lines and the pressurizer surge line. The changes would permit the licensee to exclude consideration of the dynamic effects associated with postulated rupture of the analyzed segments of the accumulator lines piping system and the pressurizer surge line piping system from the current licensing basis.

2.0 REGULATORY EVALUATION

The Nuclear Regulatory Commission (NRC) staff finds that the licensee in its September 30, 2004, and May 28, 2005, letters, addressed the applicable regulatory requirements. The regulatory requirements upon which the NRC staff based its review of the application are as follows:

1. Appendix A, "General Design Criterion" (GDC), to Part 50 of Title 10 of the *Code of Federal Regulations* (10 CFR) establishes the minimum design requirements for the principal design criteria of water-cooled nuclear power plants. In this regard, these requirements include GDC 4, "Environmental and dynamic effects design bases." GDC-4 states in part that structures, systems, and components important to safety shall be designed to accommodate the effects of and to be compatible with the environmental conditions associated with postulated accidents. However, dynamic effects associated with postulated pipe ruptures may be excluded from the design basis when analyses reviewed and approved by the NRC demonstrate that the probability of fluid system piping rupture is extremely low under conditions consistent with the design basis for the piping.

2. Ginna UFSAR Section 3.6, "Protection Against the Dynamic Effects Associated with the Postulated Rupture of Piping," describes the design features of the Ginna Station that

protect essential equipment from the consequences of postulated piping failures both inside and outside containment. The UFSAR also states that the analyses showed that, with certain modifications proposed by the licensee, 10 CFR Part 50, Appendix A, GDC 4 was met, in that all structures, systems, and components are designed to accommodate the effects of and are compatible with the environmental conditions associated with MODES 1 and 2, maintenance, testing, and postulated accidents, including loss-of-coolant accidents.

3. NRC Draft Standard Review Plan (SRP) Section 3.6.3, "Leak-Before-Break Evaluation Procedures," dated August 28, 1987, and NUREG-1061, Volume 3, "Report of the U.S. Nuclear Regulatory Commission Piping Review Committee, Evaluation of Potential for Pipe Breaks," dated November 1984, provide guidance on the LBB method approach.

3.0 TECHNICAL EVALUATION

The LBB analysis for the pressurizer surge line was prepared by Westinghouse for the licensee and was documented in Westinghouse Report WCAP-16311-P (proprietary), Revision 0, "Technical Justification for Eliminating Pressurizer Surge Line Rupture as the Structural Design Basis for the R. E. Ginna Nuclear Power Plant," and WCAP-16311-NP (non-proprietary). The LBB analysis for the accumulator lines was prepared by Structural Integrity Associates, Inc., for the licensee and was documented in its Report SIR-99-036, Revision 0, "Leak-Before-Break Evaluation of Portions of the Accumulator A and B Piping at R. E. Ginna Nuclear Power Station."

3.1 Pressurizer Surge Line

The 10-inch pressurizer surge line connects reactor coolant system (RCS) hot-leg B to the bottom of the pressurizer. The line is run along the loop B compartment wall and an exterior vertical wall of the refueling canal before turning upward to connect to the bottom of the pressurizer. Rupture of the line may require operation of the nearby low-pressure safety injection, high-pressure safety injection, and containment spray to mitigate the loss-of-coolant accident. These lines, although nearby, are mostly routed on the underside of the refueling canal which is above the basement floor. The surge line and mitigating equipment pipes are on walls which are normal to each other at an exterior corner over most of the pipe run.

3.1.1 Scope of the LBB Evaluation

The scope of the LBB evaluation for the pressurizer surge line covers the entire line from the primary loop nozzle junction to the pressurizer shell nozzle. The surge line was fabricated from wrought austenitic stainless steel, American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code) material classification SA-376, Type 316, and does not include any cast materials.

The piping welds were fabricated from stainless steel using gas tungsten arc welding (GTAW) and/or shielded metal arc welding (SMAW). None of the welds contain Alloy 82/182 material; therefore, primary water stress-corrosion cracking (PWSCC) is not a concern for the welds. There is a stainless steel safe-end piece between the nozzle and reducer at the pressurizer nozzle location. The outside diameter of the pipe is 10.75 inches and the minimum wall thickness is 0.896 inch.

The NRC staff finds that the licensee adequately defined the analyzed segments of the piping system for which LBB approval was sought.

3.1.2 LBB Methodology

Draft SRP 3.6.3 and NUREG-1061, Volume 3, specify that the LBB approach should not be applied to high energy piping that has experienced stress-corrosion cracking, water hammer or low- and high-cycle fatigue. The licensee established that no active degradation mechanisms (e.g., flow accelerated corrosion, stress-corrosion cracking, fatigue) were expected in the subject piping segments. The licensee established that no unanalyzable loading events (water hammer) would be expected to occur in the surge piping segments. The licensee also stated that there has been no service-induced cracking or wall thinning in the surge lines of Westinghouse PWRs.

On April 29, 2005, the NRC staff asked the licensee for information about the operating experience of the surge line at Ginna. In its May 28, 2005, response, the licensee stated that as part of the Ginna inservice inspection program, welds in the pressurizer surge line are periodically inspected. During the 2003 refueling outage, based on the liquid penetrant test, the licensee found indications on the outer diameter surface of the weld that connects the safe-end to the pressurizer surge nozzle. A boat sample was removed from a section containing a number of indications and examined to determine the root cause of the indications. The root cause was attributed to hot cracking, which developed during original construction. Similar indications were observed during the 2005 refueling outage inspection of the same weld. The indications were ground out until the linear indications disappeared. It appears that the indications were not service-induced and not caused by any active degradation mechanism.

The mechanical properties of the surge line at room temperature were obtained from the manufacturer’s certified materials test reports. The minimum and average tensile properties were calculated by using the ratio of the ASME Code, Section II properties at various temperatures. The representative minimum yield strength and minimum ultimate strength at operating temperature were used for the flaw stability evaluations and the representative average yield strength was used for the leak rate predictions. The NRC staff finds that the material property parameters used by the licensee were consistent with Draft SRP 3.6.3.

Based on consideration of the highest stress locations coincident with the worst material properties, the licensee identified three bounding locations at nodes 1020, 1120, and 1280 for the LBB analysis. Node 1020 has the highest stress and is located at the weld joint between the surge line and the hot leg. Node 1120 has the second highest stress and is located at the weld at the end of the bend of the pipe. Node 1280 has the third highest stress and is located at the weld joint between the surge line and pressurizer nozzle.

At the three limiting locations, the licensee calculated leakage flaw sizes using loading associated with normal operating conditions, which included axial forces and moments due to pressure, dead weight, and thermal expansion. The licensee calculated the length of a through-wall circumferential flaw at the three weld locations that would generate a leakage rate of 2.5 gallons per minute (gpm). This evaluation was based on the crack morphology parameters (surface roughness and number of turns) associated with fatigue cracks. Comparing a leak rate of 2.5 gpm to a detection capability of 0.25 gpm within 1 hour, the licensee achieved a margin of 10 and, thus, satisfies the LBB criterion in Draft SRP 3.6.3.

In recent industry experience, improved fuel integrity and reduced RCS radioactivity levels have caused the gaseous channel of the containment atmosphere radiation monitor to become less effective for RCS leakage detection. The detection of RCS leakage could take longer than is required in the plant technical specifications. In light of the experience, the NRC staff asked the licensee whether the current leakage detection capability of 0.25 gpm at Ginna can still be maintained and satisfy Regulatory Guide (RG) 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems," May 1973. The licensee responded that it is aware of recent industry experience in the leakage detection systems and the leakage detection systems; however, Ginna has demonstrated conformance with RG 1.45. To support its argument, the licensee cited its previous LBB application submittal and associated staff review for the residual heat removal (RHR) system piping in 1998. By a letter dated August 8, 1998, the licensee responded to the NRC staff request for additional information related to its LBB application to the RHR system piping. In that response, the licensee stated that the particulate monitors were demonstrated to be capable of detecting very small leak rates, even with robust fuel. The licensee did not credit the gaseous monitors to meet the 0.25 gpm within 1 hour detection capability, although the monitors are a useful backup. The second credited leak detection system is inventory balance. In its February 25, 1999, letter to the licensee, the NRC staff approved the LBB application for the RHR system piping. In the NRC staff's safety evaluation, the staff found that the 0.25 gpm detection capability is acceptable because Ginna has a relatively small containment volume, effective recirculation of air in the containment, and the second generation of R-11 detector. For the current LBB application on the surge line, the staff finds that the leakage detection capability of 0.25 gpm is acceptable based on the previous information submitted.

The licensee calculated the critical flaw sizes for the 3 bounding locations based on limit load analysis, which follows the net section collapse criterion in NUREG-1061, Volume 3. The licensee used the loading from the faulted conditions, which include normal operating conditions in conjunction with safe shutdown earthquake and seismic anchor motion loads. The severe transients such as thermal stratification and forced cooldown (also known as pressurizer reflood) were included. In the critical flaw calculation, the "absolute sum" method was used to add the individual axial forces and moments into the combined axial forces and moments. When analyzing the stainless steel weld using a limit load approach, an additional factor (Z-factor) was incorporated to account for the generally lower toughness and low load carrying capacity of the SMAW welds. The licensee applied the Z-factor to increase the applied loads and thus reduce the critical flaw size, which would be conservative. The ratios between the critical flaw size and the leakage flaw size for the bounding locations maintained a factor of 2, which satisfied the guidance in Draft SRP 3.6.3.

The licensee performed a fatigue crack growth analysis to determine the sensitivity of the surge line to the postulated small cracks when subjected to the various transients. Five cracks were assumed at the reducer between the surge line pipe and pressurizer nozzle. The initial flaws were assumed to be 10% of the wall thickness with an aspect ratio (crack length to depth) of 6 to 1. The result showed that the maximum final crack size after 60 years was insignificant. The flaw growth through the pipe wall is not expected to occur and the licensee concluded that fatigue crack growth is not a concern.

The NRC staff confirms that the surge line can be shown to exhibit LBB behavior consistent with the guidance in Draft SRP 3.6.3 and NUREG-1061, Volume 3. The licensee has shown that: (1) a margin of 10 exists between the calculated leak rate from the leakage flaw size and the detection capability of the leakage detection system; (2) a margin of 2 or more exists between the critical flaw size and the leakage flaw size having a leak rate of 2.5 gpm; (3) loadings are applied to postulated cracks and pipes consistent with SRP 3.6.3; (4) fatigue crack growth in 60 years has been shown to be insignificant; and (5) there are no active degradation mechanisms associated with the surge line.

3.2 Accumulator Lines

The 2 safety injection accumulators (A and B) function passively to discharge at least 2100 parts per million (ppm) borated water into the cold legs of the RCS piping, thus ensuring immediate core cooling. To provide protection for large area ruptures in the RCS, the safety injection system must respond to rapidly reflood the core following the rapid depressurization and core voiding that is characteristic of large area ruptures. The accumulators act to perform the rapid reflooding function with no dependence on normal or emergency power sources, and also with no dependence on receipt of an actuation signal. The accumulator piping considered in the LBB evaluation is adjacent to the RCS cold legs.

3.2.1 Scope of the LBB Evaluation

The ASME Code Class 1 portion of the accumulator A piping extends from the RCS loop cold leg nozzle to check valve 867A and motor-operated valve 721 (Nodes 856 through 960). This portion of the accumulator A piping also serves as part of the RHR system. All the nodal locations on the affected portion with the exception of Node 856 were considered in the LBB evaluation of the RHR system piping which the staff approved in a safety evaluation dated February 25, 1999. Therefore, the scope of the LBB evaluation for the accumulator A line in the licensee’s application involves a short pipe segment that includes only Node 856, which is located at one end of check valve 867A.

The scope of the LBB evaluation for the accumulator B line includes only the elbow between the cold leg nozzle of the RCS loop (Node 60) to check valve 867B (Node 80), excluding the valve itself. There is no flow in the accumulator lines during normal operation, including no possibility of cold in-leakage past the isolation valves, and complex system transients are not involved.

The accumulator piping for both A and B is constructed from ASME Code, Section II, material classification SA-376, Type 316, stainless steel. The welds are fabricated with stainless steel electrodes (ASME IX filler material E316) using SMAW processes. The licensee stated that the welds in both accumulator lines do not contain Alloy 82/182 material, which is susceptible to stress-corrosion cracking. The piping is schedule 160 with a nominal diameter of 10 inches. The operating pressure is 2235 psig and the operating temperature is 550 °F.

The NRC staff finds that the licensee adequately defined the analyzable segments of the piping system.

3.2.2 LBB Methodology

Draft SRP 3.6.3 and NUREG-1061, Volume 3, specify that the LBB approach should not be applied to high energy piping that has experienced stress-corrosion cracking, water-hammer, or, low- and high-cycle fatigue. The licensee established that no active degradation mechanisms (flow accelerated corrosion and stress-corrosion cracking) were expected in the accumulator piping segments. The licensee referenced an NRC report, NUREG-0927, in which it is stated that the probability of water hammer occurrence in the affected portions of the accumulator system is very low. In addition, the licensee has implemented Electric Power Research Institute (EPRI) guidelines TR-106438, “Water Hammer Handbook for Nuclear Plant Engineers and Operators,” May 1996, to prevent and mitigate water hammer events at Ginna. As for fatigue, the licensee demonstrated by fatigue crack growth analysis as discussed below that fatigue will not be a significant problem for the accumulator lines.

NUREG-1061, Volume 3, recommends that actual plant-specific material properties be used in the LBB evaluations. The licensee stated that actual archival materials for the accumulator piping is not available. Hence, the licensee used the least favorable material properties from the EPRI Ductile Fracture Handbook, 1989, which was also the basis for flaw acceptance criteria in the ASME Code, Section XI.

The material properties of interest for fracture mechanics and leakage calculations are the modulus of elasticity, the yield stress, the ultimate stress, the Ramberg-Osgood parameters for describing the stress strain curve, the fracture toughness, and the power law coefficient for describing the material J-Resistance curve. In the analysis, the least favorable of the base metal and weld metal properties were used to obtain conservative results.

Considering the highest stress locations coincident with the worst material properties, the licensee identified the following limiting locations for the LBB analysis. For the accumulator A pipe, the critical location is the weld between the pipe and check valve 867A (Node 856). For the accumulator B pipe, the two critical locations are the weld joint between the accumulator B pipe and the cold leg nozzle (node 60), and the weld joint between the accumulator pipe elbow and check valve 867B (node 80). These nodes were considered for the analysis because they are located at welds in a tee and elbow and consequently reflect high stresses due to the stress intensification effects. In addition, the SMAW weld properties at these nodes will provide the most conservative critical flaw and leakage flaw sizes because of its low toughness and susceptibility to thermal aging.

At the three critical locations, the licensee calculated the leakage flaw sizes using loading associated with normal operating conditions, including axial forces and moments due to pressure, dead weight, and thermal expansion. The licensee calculated the length of a through-wall circumferential flaw which would generate a leakage rate of 2.5 gpm, which is 10 times the leakage detection capability of 0.25 gpm at Ginna.

The licensee calculated the critical flaw sizes at the critical locations using elastic-plastic fracture mechanics with the J-integral method. The pipe loading applied to the cracks were axial forces and moments of normal operating and faulted conditions, including safe shutdown earthquake and seismic anchor motion loads. The licensee used the “absolute sum” method to add the individual axial forces and moments into the combined axial forces and moments.

The licensee performed a fatigue crack growth analysis to determine the sensitivity of the accumulator lines to the postulated small cracks when subjected to the various transients. Cracks of various depth and aspect ratios were assumed. The licensee’s fatigue crack growth analysis showed that in 60 years the fatigue crack growth is insignificant. The analysis also showed that the crack will grow through-wall before extending in length significantly. This indicates that leakage will occur before safety margins are exceeded.

The NRC staff confirms that the proposed pipe segments in accumulator lines A and B can be shown to exhibit LBB behavior consistent with the guidance in Draft SRP 3.6.3 and NUREG-1061, Volume 3. The licensee has shown that: (1) a margin of 10 exists between the calculated leak rate from the leakage flaw and the leak detection capability of 0.25 gpm; (2) a margin of 2 or more exists between the critical flaw and the flaw having a leak rate of 2.5 gpm; (3) fatigue crack growth in 60 years has been shown to be insignificant; (4) loadings are applied to postulated cracks and pipes consistent with SRP 3.6.3; and (5) there are no active degradation mechanisms associated with accumulator lines A and B.

3.3 Staff Evaluation

The NRC staff finds that the licensee has demonstrated that the pressurizer surge line and the proposed pipe segments in accumulator lines A and B exhibit LBB behavior consistent with the guidance in Draft SRP 3.6.3 and NUREG-1061, Volume 3. Therefore, the staff concludes that, in accordance with GDC 4 of Appendix A to 10 CFR Part 50, the licensee is permitted to exclude consideration of the dynamic effects associated with the postulated rupture of the analyzed segments of the pressurizer surge line piping system and the postulated rupture of the analyzed segments of the accumulator piping system from the current licensing basis at Ginna.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the New York State official was notified of the proposed issuance of the amendment. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding (70 FR 38721). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

6.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above that (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributor: J. Tsao

Date: September 22, 2005