March 8, 2010

Mr. John T. Carlin Vice President
RE. Ginna Nuclear Power Plant
RE. Ginna Nuclear Power Plant, LLC
1503 Lake Road
Ontario, NY 14519

SUBJECT: RELIEF REQUEST NUMBER 24 - FOURTH INTERVAL INSERVICE INSPECTION PROGRAM PROPOSED ALTERNATIVE FOR BOTTOM MOUNTED INSTRUMENTATION EXAMINATIONS - R.E. GINNA NUCLEAR POWER PLANT (TAC NO. ME1364)

Dear Mr. Carlin:

By letter dated May 22, 2009, as supplemented by letter dated August 14, 2009, R.E. Ginna Nuclear Power Plant, LLC, the licensee for the R.E. Ginna Nuclear Power Plant, requested Nuclear Regulatory Commission (NRC) approval to use an alternative to the requirements of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code, Section XI, Code Case N-722 for the bare metal visual inspection of the bottom mounted instrument (BMI) nozzles during the fourth 10-year inservice inspection (ISI) interval. Specifically, the alternative was requested pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.55a(a)(3)(i).

The NRC staff has completed its review of the information provided by the licensee for Relief Request No. 24. The staff concludes that the information provided by the licensee supports the granting of an alternative pursuant to 10 CFR 50.55a(a)(3)(i) because the alternative provides an acceptable level of quality and safety and reasonable assurance of continued structural integrity for the Ginna BMI nozzles during the fourth 10-year ISI interval. Verbal authorization of the licensee's proposed alternative was provided on August 19, 2009.

Sincerely,

Nancy L. Salgado

Nancy L. Salgado, Chief
Plant Licensing Branch I-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-244

Enclosure:
Safety Evaluation

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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO RELIEF REQUEST NO. 24

RENEWED FACILITY OPERATING LICENSE NO. DPR-18

R. E. GINNA NUCLEAR POWER PLANT, LLC

R. E. GINNA NUCLEAR POWER PLANT

DOCKET NO. 50-244

1.0 INTRODUCTION

By letter dated May 22, 2009 (Agencywide Document Access and Management System (ADAMS) Accession No. ML091530248), as supplemented by letter dated August 14, 2009 (ADAMS Accession No. ML092310543), R.E. Ginna Nuclear Power Plant, LLC (Ginna, the licensee), submitted a request for authorization of an alternative to certain requirements of the plant's fourth 10-year Inservice Inspection (ISI) interval. The request for authorization of the alternative was made pursuant to the provisions of Title 10 of the *Code of Federal Regulations* (10 CFR) 50.55a(a)(3)(i). Specifically, the licensee proposed an alternative to the bare metal visual examination requirements of 10 CFR 50.55a(g)(6)(ii)(E). The licensee requested approval of this alternative for the examinations required during its refueling outage scheduled to begin in September 2009. Verbal authorization of the licensee's proposed alternative was provided on August 19, 2009 (ADAMS Accession No. ML092320057).

2.0 REGULATORY EVALUATION

ISI of American Society of Mechanical Engineers (ASME) Code Class 1, 2, and 3 components is performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code (ASME Code) and applicable addenda as required by 10 CFR 50.55a(g), except where specific relief has been granted by the Nuclear Regulatory Commission (NRC) pursuant to 10 CFR 50.55a(g)(6)(i). 10 CFR 50.55a(a)(3) states that alternatives to the requirements of paragraph (g) may be used, when authorized by the NRC, if: (i) the proposed alternatives would provide an acceptable level of quality and safety or (ii) compliance with the specified requirements would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

Pursuant to 10 CFR 50.55a(g)(4), ASME Code Class 1, 2, and 3 components shall meet the requirements, except the design and access provisions and the preservice examination requirements, set forth in the ASME Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," to the extent practical within the limitations of design, geometry, and materials of construction of the components. The regulation requires that inservice examination of components and system pressure tests conducted during the first 10-year interval and subsequent intervals comply with the requirements in the latest edition and addenda of Section XI of the ASME Code incorporated by reference in 10 CFR 50.55a(b) twelve months prior to the start of the 120-month interval, subject to the limitations and modifications listed therein. The applicable Code of record for the fourth 10-year interval ISI program at Ginna is the 1994 Edition of the ASME Code, Section XI, with 1996 addenda. The fourth 10-year interval ISI program at Ginna ended on December 31, 2009.

In addition, the NRC identifies augmented examination requirements in 10 CFR 50.55a. One such requirement relevant to this licensee request is found in 10 CFR 50.55a(g)(6)(ii)(E). 10 CFR 50.55a(g)(6)(ii)(E) requires that pressurized-water reactor (PWR) licensees implement the requirements of ASME Code Case N-722, "Additional Examinations for PWR Pressure Retaining Welds in Class 1 Components Fabricated With Alloy 600/82/182 Materials, Section XI, Division 1." This augmented inspection requirement applies to the reactor vessel bottom mounted instrumentation (BMI) nozzles at Ginna.

3.0 TECHNICAL EVALUATION

3.1 Component for which the Proposed Alternative is Requested

The licensee has submitted the proposed alternative for the ASME Code Class 1 BMI nozzles on the Ginna reactor vessel.

3.2 ASME Code/10 CFR 50.55a Requirement

ASME Code Case N-722 is mandated by NRC as an augmented ISI requirement in 10 CFR 50.55a(g)(6)(ii)(E) and requires that bare metal visual examinations be performed on reactor vessel BMI nozzles every other refueling outage.

3.3 Licensee's Proposed Alternative

The licensee's proposed alternative consists of an ultrasonic testing (UT) examination from inside the nozzles in 2011 of each BMI nozzle above, over, and below the partial penetration weld between the penetration nozzle and the reactor vessel. The proposed alternative also includes visual examination of each of the 36 nozzle locations from outside of the reactor vessel each refueling outage. Ten of the nozzle locations are completely visually occluded by an originally installed coating. The remaining nozzles are all at least partially occluded by the originally installed coating. Ginna proposes to perform a detailed visual examination in 2009 and again in 2011 without skipping an outage. Approval of the licensee's alternative examination would permit the licensee to perform the next required inspections in the fall of 2009.

3.4 Licensee's Basis for Proposed Alternative

The licensee's basis for requesting the proposed alternative noted that an evaluation of the coating information by a coatings specialist determined that if a reactor coolant system (RCS) leak developed through the BMI nozzle weld or base material, the coating would be unlikely to resist RCS pressure and would instead either blister or pass leakage that could be detected by visual examination. The licensee's evaluation assumed that the load experienced by the coating if the annulus filled with water at RCS pressure would be approximately 5 pounds (because RCS pressure would be applied to a very small area). The licensee's evaluation concluded that the load developed by RCS pressure should be high enough to overcome adhesive and cohesive strength of the coating causing a thin layer of coating to blister. Ginna also indicated that the visual examination procedure they intend to implement has a higher visual resolution requirement than that required in ASME Code Case N-722.

The use of UT examination of the BMI nozzle base material will be accomplished using procedures, equipment and personnel who demonstrate proficiency in detecting flaws in site-specific mockups. Ginna concluded that a UT examination will identify potential flaws in the BMI nozzle base material whose leakage could otherwise be masked by the coating during the visual examination. Additionally, Ginna noted that the UT examination could identify precursors to leakage by finding part-through-wall cracks, if any exist, in the base material. Ginna provided an evaluation of service experience with partial penetration nozzles manufactured of Alloy 600 and noted that cracks in the base material are much more likely than cracks in weld metal. Therefore, the licensee concluded that the use of the UT and visual inspection techniques will result in detection of flaws that could lead to potential leakage that would otherwise be detected by the bare metal visual inspection of the BMI nozzles required by 10 CFR 50.55a(g)(6)(ii)(E).

3.5 NRC Staff Evaluation

Primary water stress-corrosion cracking (PWSCC) in J-groove welded Alloy 600 nozzles has been observed in a variety of applications. The safety issue associated with such cracking is that the presence of water on the outside diameter (OD) of the nozzle has led to initiation of circumferential cracks on the OD of reactor vessel head penetrations at other plants. Circumferential cracking creates the possibility of ejection of the nozzle and a consequential loss-of-coolant accident. Axial cracks in the nozzle and cracks through the weld do not, by themselves, present immediate safety issues. This is because axial cracks are supported by the surrounding vessel material which prevents rupture until they grow to a significant distance outside the reactor vessel head. The residual stresses that drive axial cracks diminish with distance from the J-groove weld, so axial cracks are predicted to never grow long enough to extend outside the annulus. With respect to cracks in the weld, registry between the PWSCC fracture surfaces is sufficient to withstand system pressure, so even J-groove welds that are cracked 360 degrees will not result in ejection of a nozzle; the attached weld nuggets prevent nozzle ejection. Therefore, the principle safety issue with respect to PWSCC is circumferential cracking that can initiate and grow on the nozzle OD.

Another potential safety issue with respect to PWSCC is boric acid corrosion of the reactor vessel head. If the nozzle annulus is not completely occluded, visual inspection at the periodicity prescribed in ASME Code Case N-722 would detect the leakage before boric acid corrosion could create a challenge to structural integrity. For leakage into an occluded annulus, the contained reactor coolant would have low oxygen levels typical of RCS chemistry. Boric acid corrosion rates are very low in deaerated water, so leakage into an occluded annulus would not result in any significant boric acid corrosion. As a result, provided visual inspections are performed periodically, boric acid corrosion is not a safety issue for BMIs.

The licensee proposes to perform a visual examination every outage, rather than every other outage, without removing the paint on the reactor vessel bottom head and also proposes to supplement the visual examination with a qualified UT examination of the nozzles during the unit's 2011 refueling outage. The NRC staff understands that Ginna is not able to perform a complete bare metal visual examination as defined in ASME Code Case N-722 due to paint that bridges over some of the nozzle annuli. The licensee indicated that they have prepared mockups in an attempt to qualify a coating removal process, but that the process has not been consistently successful, and its application in the field could lead to unnecessary radiation exposure. The extent of paint occlusion of the nozzle annuli (where paint is in the gap between the nozzle OD and the nozzle bore in the vessel) has been documented and ranges from 12.5% to 100%, with 10 nozzles being 100% occluded. The licensee indicates that their alternative provides an acceptable level of quality and safety.

For nozzles that have annuli occluded less than 100%, the NRC staff believes it is highly likely that any leakage of borated water through the nozzle wall or through the J-groove weld would flow down towards the paint, and then would flow through a portion of the annulus that was not occluded. Evidence of leakage would be visible as boric acid deposits. The concern with partially occluded nozzles is that certain leaks may take longer to reach the outside of the annulus than would be the case if the annulus were not occluded at all. ASME Code Case N-722, which the NRC endorses, set the acceptable time between inspections as every other refueling outage, which can be as long as 4 years. The licensee's proposal to perform the examination every outage compensates for the extra time that might be required for evidence of leakage to become visually discernable on partially occluded nozzles.

For nozzles that are 100% occluded, or nearly 100% occluded, the concern is that the coating could retain any leakage inside the annulus, preventing visual examination from being an effective technique for discovering cracked nozzles. The NRC staff reviewed the licensee's evaluation, which treated the coating material as a typical thin coating film. The staff agrees that a thin film would be very likely to rupture, and would in any event blister, if subjected to RCS pressure. Therefore, visual examination will provide information about whether PWSCC of the BMI nozzles is occurring. The visual examination will identify the presence or absence of leakage on nozzles that are significantly less than 100% occluded and, for those nozzles that are significantly occluded but that have thin films occluding the annulus, examination for blistering will provide evidence of the presence or absence of PWSCC leakage.

However, the NRC staff has postulated that when the coating was originally applied it could have been drawn by capillary action up into the nozzle annulus, potentially forming a thick plug in the annulus. The staff recognizes that the formation of a thick plug would have been dependent on the viscosity of the uncured coating and the application techniques used to apply the coatings. No information related to these application variables was available. If a thick plug of coating material did form, the area of the coating exposed to RCS pressure in the event of a leak could be backed by a centimeter or more of coating thickness which would be restrained and reinforced by the adjacent reactor vessel shell and nozzle material. Once cured, the coating could plug the annulus if the strength of the coating bond and the shear strength of the coating plug were sufficiently high. The licensee was unable to produce data that demonstrated the cohesive or adhesion strength of the coating material, so was unable to demonstrate that RCS pressure would provide sufficient force to break through a coating plug. Without specific property information on the coating, the staff cannot conclude that a thick plug similar to what may exist in an occluded annulus would rupture or blister if it were exposed to RCS pressure inside the annulus. The licensee indicated they would attempt to retrieve some coating material during the fall 2009 refueling outage and, if successful, would attempt to develop quantitative information related to the coating properties. Such information may enable the licensee to develop alternatives for their longer term plans for inspecting BMI nozzles.

To address the issue of potential OD initiated circumferential cracking, Ginna will perform UT of the nozzles during the 2011 outage. Ultrasonic examination will be able to detect axial cracking in the nozzle material that could be a precursor to OD cracking. Additionally, UT examination will detect any OD initiated cracking that was caused by reactor coolant that leaked through the J-groove weld. The NRC staff considers the UT examination to provide an acceptable level of quality and safety as compared to the visual examination.

Further, Ginna committed to simulate a RCS break location at the bottom of the reactor vessel and determine if significant differences in operator response for a bottom of reactor vessel break and a traditional cold leg break exist. If significant differences are identified, then Ginna committed to schedule additional simulator training for the unit's operators during the first training cycle following startup from the fall 2009 refueling outage.

The NRC staff concludes that the proposal to perform visual examination every refueling outage provides acceptable safety and quality for addressing potential leaks from nozzles whose annuli are not completely occluded by coatings. For all other nozzles, the staff concludes that UT examination of the nozzles every other refueling outage provides reasonable assurance and an acceptable level of quality and safety in the ability to detect cracking in a timeframe that is commensurate with the timeliness of a visual examination of a non-occluded nozzle.

4.0 CONCLUSION

The NRC has completed its review of Relief Request No. 24 provided in the licensee's submittal dated May 22, 2009, as supplemented by letter dated August 14, 2009. The staff concludes that the information provided by the licensee supports the granting of an alternative pursuant to 10 CFR 50.55a(a)(3)(i) because the alternative provides an acceptable level of quality and safety and reasonable assurance of continued structural integrity for the Ginna BMI nozzles during the fourth 10-year ISI interval. Therefore, the proposed alternative is authorized pursuant to 10 CFR 50.55a(a)(3)(i) for the 2009 and 2011 refueling outage. All other requirements of the ASME Code, Section XI, for which relief has not been specifically requested and approved, remain applicable, including third party review by the Authorized Nuclear Inservice Inspector.

Principal Contributor: Robert Hardies

Date: March 8, 2010

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Mr. John T. Carlin Vice President
RE. Ginna Nuclear Power Plant
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Dear Mr. Carlin:

By letter dated May 22, 2009, as supplemented by letter dated August 14, 2009, R.E. Ginna Nuclear Power Plant, LLC, the licensee for the R.E. Ginna Nuclear Power Plant, requested Nuclear Regulatory Commission (NRC) approval to use an alternative to the requirements of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code, Section XI, Code Case N-722 for the bare metal visual inspection of the bottom mounted instrument (BMI) nozzles during the fourth 10-year inservice inspection (ISI) interval. Specifically, the alternative was requested pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.55a(a)(3)(i).

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Sincerely,
/RA/
Nancy L. Salgado, Chief
Plant Licensing Branch I-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-244
Enclosure:
Safety Evaluation

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