UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

June 29, 2012

Mr. Joseph E. Pacher
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 – RELIEF REQUEST FOR AUTHORIZATION OF A PROPOSED ALTERNATIVE FOR CERTAIN REQUIREMENTS OF INSERVICE INSPECTION PROGRAM FOR EXAMINATION OF BOTTOM MOUNTED INSTRUMENTATION NOZZLES (TAC NO. ME7731)

Dear Mr. Pacher:

By letter dated December 16, 2011(Agencywide Documents Access and Management System (ADAMS) Accession No ML11363A074), R.E. Ginna Nuclear Power Plant, LLC (licensee), submitted a request to U.S. Nuclear Regulatory Commission (NRC) for authorization of an alternative to certain requirements of the R.E. Ginna Nuclear Power Plant's (Ginna's) fifth 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 has 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 all future refueling outages.

The licensee stated that as-found visual examinations of the bottom mounted instrumentation (BMI) nozzles during every outage will provide superior certainty of condition than the code required bare-metal visual examinations during every other refueling outage. The licensee justifies this alternative by citing evidence gathered from ultrasonic examinations conducted during the 2011 Ginna outage, past visual examinations, surface evidence gathered from the BMI nozzles, and full-scale BMI mockup testing conducted with the assistance of the Southwest Research Institute.

The NRC staff has completed its review of the information provided in the licensee's submittal. Based on that review, and in accordance with 10 CFR 50.55a(a)(3)(i), the NRC has concluded, that the licensee provided adequate information regarding the proposed alternative to substantiate that its implementation of the alternative proposed in the submittal will provide an acceptable level of quality and safety. Therefore, the proposed alternative is authorized pursuant to 10 CFR 50.55a(a)(3)(i) for Ginna for the fifth ISI interval. The applicable Code of record for the fifth 10-year interval ISI program at Ginna is the 2004 Edition of the ASME Code, Section XI, with no addenda. The fifth 10-year interval ISI program at Ginna will end in December, 2019.

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.

If you have any questions, please contact Mohan Thadani, the NRC's Project Manager for Ginna at (301) 415-1476 or email mohan.thadani@nrc.gov.

Sincerely,

RGuzman for,

George Wilson, Chief
Plant Licensing Branch I-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-244

Enclosure:
As stated

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELIEF REQUEST NO ISI-05

FIFTH 10-YEAR INSERVICE INSPECTION INTERVAL

R.E. GINNA NUCLEAR POWER PLANT

R.E. GINNA NUCLEAR POWER PLANT, LLC

DOCKET NO. 50-244

1.0 INTRODUCTION

By letter dated December 16, 2011, (Agencywide Documents Access and Management System (ADAMS) Accession No. ML11363A074), R.E. Ginna Nuclear Power Plant, LLC (licensee), submitted a request to U.S. Nuclear Regulatory Commission (NRC) for authorization of an alternative to certain requirements of the Ginna Nuclear Power Plant's (Ginna's) fifth 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 has 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 all future refueling outages.

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 Code and applicable addenda as required by Title 10 of the *Code of Federal Regulations* (10 CFR) 50.55a(g), except where specific relief has been granted by the NRC pursuant to 10 CFR 50.55a(g)(6)(i). It states in 10 CFR 50.55a(a)(3) 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 and 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 ten-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 fifth 10-year interval ISI program at Ginna is the 2004 Edition of the ASME Code, Section XI, with no addenda. The fifth 10-year interval ISI program at Ginna will end in December, 2019.

In addition, the NRC identifies augmented examination requirements in 10 CFR 50.55a. One such requirement relevant to this request is found in 10 CFR 50.55a(g)(6)(ii)(E). It is mandated in 10 CFR 50.55a(g)(6)(ii)(E) 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 Components for which the Proposed Alternative is Requested

The licensee has submitted the proposed alternative for the 36 ASME Code Class 1 Inconel 600 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 during every other refueling outage.

3.3 Licensee's Proposed Alternative

The licensee's proposed alternative consists of "conducting as-found visual examinations of the BMI surfaces during every refueling outage to the visual requirements (VE, enhanced visual) of Code Case N-722-1."

3.4 Licensee's Basis for Proposed Alternative

The licensee proposes that as-found visual examinations of the BMI nozzles during every outage will provide superior certainty of condition than the code required bare-metal visual examinations during every other refueling outage. The licensee justifies this alternative by citing evidence gathered from ultrasonic examinations conducted during the 2011 Ginna outage, past visual examinations, surface evidence gathered from the BMI nozzles, and full-scale BMI mockup testing conducted with the assistance of the Southwest Research Institute.

The ultrasonic (UT) examination conducted in 2011 interrogated the volume of the BMI tube and weld interface to the intent of ASME Section XI Appendix VIII. The examination "did not identify ID [inner diameter] or OD [outer diameter] initiated cracking." One "original manufacturing defect" was found on BMI nozzle A86 and was analyzed according to ASME Code, Section XI, paragraph IWB-3144(b). This analysis was submitted as a separate submittal. The licensee concluded from the UT results that no service induced cracking was identified.

In 2009, the licensee took paint samples from the machined nozzle OD surface and the lower head for examination. During this process cracks in the paint were identified in the nozzle annulus. In 2011 metallurgical replicas were taken of the nozzle-to-head annulus at three locations in which paint-cracking was also identified. None of the evidence gathered indicated that the paint would serve as an obstacle to detecting leaks or leak indications. In addition, full-scale BMI mockup testing was performed with the following conclusions:

1. Paint was not found to be an effective barrier for either liquid or steam exiting the annulus under even a fraction of operating pressure and temperature,

2. Paint cracking as observed on the Ginna BMI nozzles was reproducible through thermal cycling of mockup paint samples, and these cracks served as effective leak paths,

3. Intact paint provided negligible impediment to leakage of steam at elevated temperatures,

4. Paint did not appear to alter boric acid crystal deposition due to steam leakage either in volume or location,

5. Paint did not appear to alter the boric acid wastage processes or rates within the annulus from bare-metal circumstances.

The licensee concluded from the testing that, should a leak path connect through the vessel into the BMI annulus, the paint would not provide any significant impediment to steam exiting the annulus or to leak detection via visual inspection. Further, the licensee stated that as-found visual examinations conducted since the 2003 Ginna outage identified no leakage to date. Therefore, the presence of paint in the BMI annulus does nothing to reduce the effectiveness of inspections despite the Code requirement of "bare metal" examinations.

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 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, the staff concluded in a related safety evaluation (ADAMS Accession No. ML100290926) that registry between the PWSCC fracture surfaces is sufficient to withstand system pressure. The staff stated that 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 challenge 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 required for bare metal examination, without removing the paint on the reactor vessel bottom head. The staff understands that the licensee 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 in its fourth ISI interval submittal (ADAMS Accession No. ML091530248) that 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.

For nozzles that have annuli occluded less than 100%, the staff expects that 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, sets the acceptable time between inspections as every other refueling outage, which can be as long as four 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. Most convincingly, the licensee's mockup testing demonstrated that the paint cracks very quickly when thermally cycled, as it would be in an operating reactor, and that the paint presents little if any barrier to leakage at any rate or temperature.

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 staff postulated in the fourth ISI interval submittal safety evaluation (ADAMS Accession No. ML100290926) 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. If a thick plug of coating material did form, the area of the coating exposed to reactor coolant system pressure in the event of a leak could be blocked 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's 2011 mockup experiments demonstrated that the cohesive or adhesion strength of the coating material was insufficient to produce the above postulated scenario. While the mockup paint was not an exact replica of the original paint, the staff considers that it was sufficiently similar to use as a reference. During testing, significant wicking of the mockup paint into the annulus region produced little to no impediment to leakage. Specifically, the mockup testing postulated the following scenarios:

Table 4-1 of Enclosure 3 of the submittal

Test Number	Annulus Gap (mils)	Thermal Cycling T_{max} °F, # of Cycles	Starting Condition	Mockup Temperature °F	Coolant Temperature °F	Time to Leak (seconds)	Maximum Annulus Pressure at Leak (psi)
1	1	550/2	Not tested due to excessive paint damage in thermal cycle				
1	10	550/2	Not tested due to excessive paint damage in thermal cycle				
2	1	350/6	Full annulus	66	35	0	19
2	1	-	Full annulus	350	550	8	149
2	10	350/6	Full annulus	62	63	8	5
2	10	-	Full annulus	350	550	17	52
3	1	350/6	Empty annulus	350	550	1.144	85
3	10	350/5	Empty annulus	350	550	During heating	1

In all of the mockup tests performed the paint failed to impede leakage of liquid or steam for an amount of time even fractionally equivalent to the inspection period, a period chosen to ensure that any leakage would be identified well before sufficient damage could accumulate to challenge the BMI nozzle integrity. Water or steam in contact with the paint rapidly found its way to the exterior of the nozzle. Consequently, the staff concluded that there is reasonable assurance that paint-plugged nozzles would function in a manner identical to bare-metal nozzles.

In summary, the staff concludes that the licensee's proposal to perform visual examination of the BMI surfaces, according to the VE requirements of Code Case N-722-1, during every refueling outage, provides acceptable level of safety and quality for addressing potential leaks from the subject nozzles because the visual examination will promptly identify the presence or absence of leakage on nozzles, and hence, will provide evidence of the presence or absence of PWSCC leakage.

4.0 CONCLUSION

Based on the completed review of the information provided in the licensee's submittal dated December 16, 2011, the NRC staff has concluded, in accordance with 10 CFR 50.55a(a)(3)(i), that the licensee provided adequate information regarding the proposed alternative to substantiate that its implementation of the alternative proposed in the submittal will provide an acceptable level of quality and safety. Therefore, the proposed alternative is authorized pursuant to 10 CFR 50.55a(a)(3)(i) for Ginna's fifth ISI interval.

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 Contributors: D. Widrevitz

Date: June 29, 2012

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.

If you have any questions, please contact Mohan Thadani, the NRC's Project Manager for Ginna at (301) 415-1476 or email mohan.thadani@nrc.gov.

Sincerely,

/ra/(RGuzman for)

George Wilson, Chief
Plant Licensing Branch I-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-244

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