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

May 5, 2009

Mr. John Carlin
Vice President R.E. Ginna Nuclear Power Plant
R.E. Ginna Nuclear Power Plant, LLC
1503 Lake Road
Ontario, NY 14519-9364

SUBJECT: RELIEF REQUEST NO. 23 RE: FOURTH INTERVAL ISI PROGRAM CATEGORY B-P EXAMS – 10 YEAR CLASS 1 LEAKAGE EXAM - R.E. GINNA NUCLEAR POWER PLANT (TAC NO. ME0456)

Dear Mr. Carlin:

By letter dated January 29, 2009, R.E. Ginna Nuclear Power Plant, LLC (the licensee), submitted Relief Request No. 23 relating to system pressure tests applicable to the R.E. Ginna Nuclear Power Plant (Ginna) for the fourth 10-year inservice inspection (ISI) interval. The relief request pertains to the boundary subject to test pressurization during performance of a system leakage test conducted at or near the end of the inspection interval. In lieu of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code) requirement to include all Class 1 pressure retaining components within the test boundary, the licensee has proposed an alternative to pressurize up to the inboard isolation valve which would exclude a small segment of the Class 1 pressure boundary from attaining test pressure.

The Nuclear Regulatory Commission staff finds that the licensee's proposed alternative provides reasonable assurance of structural integrity and is, therefore, acceptable. Based on the information provided in Relief Request No. 23, the staff concludes in the enclosed safety evaluation that the licensee's compliance to the ISI Code of Record would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety. Therefore, pursuant to Title 10 of the *Code of Federal Regulations,* 50.55a(a)(3)(ii), the staff authorizes the ISI program alternative proposed in Relief Request No. 23 for the fourth 10-year ISI interval for the Ginna Nuclear Power Plant.

Please contact Douglas Pickett at 301-415-1364 or Douglas.Pickett@nrc.gov if you have any questions.

Sincerely,

John P. Boska

John Boska, Acting Chief
Plant Licensing Branch I-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-244

Enclosure:
Safety Evaluation

cc w/encl: Distribution via ListServ

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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

INSERVICE INSPECTION PROGRAM RELIEF REQUEST NO. 23

R.E. GINNA NUCLEAR POWER PLANT, LLC

R.E. GINNA NUCLEAR POWER PLANT

DOCKET NO. 50-244

1.0 INTRODUCTION

By letter dated January 29, 2009, Agencywide Document Management System Accession No. ML090360492, R.E. Ginna Nuclear Power Plant, LLC (the licensee), submitted Relief Request No. 23 relating to system pressure tests applicable to the R.E. Ginna Nuclear Power Plant (Ginna) for the fourth 10-year inservice inspection (ISI) interval. The relief request pertains to the boundary subject to test pressurization during performance of a system leakage test conducted at or near the end of inspection interval. In lieu of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code) requirement to pressurize all Class 1 pressure retaining components within the system boundary, the licensee has proposed an alternative to pressurize up to the inboard isolation valve which would exclude a small segment of the Class 1 pressure boundary from attaining Code-required test pressure. The licensee proposes to conduct a visual examination during the leakage test that would include all components within the system boundary. The relief request pertains to portions of the safety injection (SI) system, shutdown cooling system, and drain lines in the reactor coolant system (RCS).

The Nuclear Regulatory Commission (NRC) staff has evaluated the licensee's request for relief pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) 50.55a(a)(3)(ii) and concluded that compliance with the requirement in the Code of Record would result in hardship without a compensating increase in the level of quality and safety.

2.0 REGULATORY REQUIREMENTS

10 CFR 50.55a(g) requires that ISI of ASME Code Class 1, 2, and 3 components be performed in accordance with Section XI of the ASME Code and applicable addenda, except where specific written relief has been granted by the Commission pursuant to 10 CFR 50.55a(g)(6)(i). According to 10 CFR 50.55a(a)(3), alternatives to the requirements of paragraph 50.55a(g) may be used, when authorized by the NRC, if an applicant demonstrates that the proposed alternatives would provide an acceptable level of quality and safety or if the specified requirement would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

Enclosure

Pursuant to 10 CFR 50.55a(g)(4), ASME Code Class 1, 2, and 3 components (including supports) 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 (ISI) of Nuclear Power Plant Components," to the extent practical within the limitations of design, geometry, and materials of construction of the components. The regulations require that ISI 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) 12 months prior to the start of the 120-month interval subject to the limitations and modifications listed in paragraph (b) of this section. The ISI Code of Record for the fourth 10-year interval at Ginna is the 1995 Edition of the ASME Code, Section XI with the 1996 Addenda.

3.0 TECHNICAL EVALUATION

System/Component(s) for Which Relief is Requested

Examination Category	Item No.	Description
B-P	B15.50	Piping – Pressure retaining boundary
	B15.70	Valves – Pressure retaining boundary

ASME Code Requirements

Table IWB-2500-1, Examination Category B-P, Item Numbers B15.50 (Piping) and B15.70 (Valves) requires that a system leakage test be conducted once each 10-year inspection interval in accordance with the requirements of IWB-5222. The pressure retaining boundary during the system leakage test conducted at or near the end of each inspection interval shall extend to all Class 1 components within the system boundary.

Licensee's Request for Relief

Relief is requested from the specific ASME Code, Section XI, Subparagraph IWB-5222(b) requirement to extend the pressure boundary to all Class 1 pressure retaining components during the system leakage tests conducted at or near the end of inspection interval. The segment of Class 1 piping between the inboard isolation valve and the outboard isolation valve/closure device including the valves and components in the system boundary will be visually examined for evidence of past leakage and/or leakage during the system leakage test conducted with the isolation valves in the position required for normal reactor startup.

Licensee's Basis for Requesting Relief (as stated in Relief Request No. 23)

> Pressurization of components above their normal alignment at normal operating temperature and pressure in order to detect leakage during the VT-2 visual examination is not necessary. Piping with two isolation valves/closure devises is designed to operate with the first isolation valve closed. Piping between the inboard isolation valve and the outboard isolation valve/closure device during normal operating pressure and temperature is normally pressurized, but at a lower pressure.

The temperatures and pressures present in Class 1 components during a system leakage test at a pressure associated with normal system operation is sufficient to qualify as a System Pressure Test. The pressure boundary integrity of these components is validated and documented using identical VT-2 visual examination requirements each refueling outage. The requested relief will apply VT-2 inspections of the Class 1 boundary beyond the first isolation valves at a stabilized pressure based on seat leakage from the first isolation valve.

Ginna LLC performs other surveillance (i.e. Local Leakage Rate Tests, Contaminated Leakage Rate Tests, Pressure Isolation Check Valve Leak Tests and Inservice Leak Rate Tests) to monitor these components for leakage. Leakage is identified using normal operating temperatures and pressure conditions. In addition to leakage testing, boric acid inspections performed during refueling outages will also identify leakage from these components.

The following specific lines are included in the request for relief.

(a) Double Isolation Valve Segments including Gates, Checks, and Globes of the RCS.

These piping segments are between an inboard isolation valve and an outboard isolation valve in the system boundary that provides double isolation of the RCS.
The isolation valves associated with double valve isolation segments are located inside containment. For a list of specific components see Attachment 1, Group A of the licensee's Relief Request dated January 29, 2009.

(b) Vent, Drain, and Test Connection Double Isolation Segments of the RCS

These are segments of piping between an inboard isolation valve and an outboard isolation closure device in the system boundaries that provides double-isolation to the RCS. The inboard isolation valves associated with vents, drains and test connection segments are located inside containment. For a list of specific components see Attachment 1, Group B of the licensee's Relief Request dated January 29, 2009.

In pressurizing the piping segments including the valves to the ASME Code-required test pressure, the licensee would be subject to hardship or unusual difficulty without a compensating increase in the level of quality and safety as stated by the licensee below:

- Special valve lineups and/or the use of temporary high pressure hoses/piping containing RCS pressure required for these tests add unnecessary challenges to the system configuration.
- The associated components and piping are located inside containment. Tests performed inside the radiologically restricted area increases the total exposure to plant personnel while modifying and restoring system lineups, as well as contamination of test equipment.
- Use of single valve isolation from systems with lower design pressure could result in over-pressurization of these systems and damage to permanent plant equipment.

- Pressurization of some double valve isolation pipe segments would require the use of temporary high pressure hoses/piping containing RCS pressure or hydrostatic test pressure hoses. These hoses would run throughout containment and is a significant personnel safety hazard should they burst and may also damage permanent plant equipment. Hoses on the floors are also a tripping hazard for all workers in containment.
- Use of a single closure device past the first isolation valve is a significant personnel safety hazard and may damage permanent plant equipment.
- Leakage past isolation valves to the RCS during special tests could affect the RCS boron concentration and complicate the task of maintaining homogeneous boron concentrations.

Licensee's Proposed Alternative

The licensee proposes an alternative method for the pressurization boundary for specified Class 1 piping. The proposed method will leave the barriers intact for the visual examination rather than opening or bypassing the first isolation barrier prior to the examination. This modified approach will result in significant personnel exposure savings as well as minimizing the risk of personnel injury or contamination associated with opening or bypassing these normally closed isolation devices. Since these pressure tests are performed at the end of refueling outage, elimination of the requirement to open or bypass these isolation devices will also minimize the impact on the outage duration.

The segments of Class 1 piping between the inboard isolation valve and outboard isolation valve/closure device including the valves/closure devices and components in the system boundary will be visually examined (VT-2) for evidence of past leakage and/or leakage during the system leakage test conducted with the isolation valves/closure devices in the position required for normal reactor start up.

4.0 STAFF EVALUATION

The Code of Record, 1995 Edition to the ASME Code with the 1996 Addenda, Section XI, Table IWB-2500-1, Category B-P, Item numbers B15.50 (Piping) and B15.70 (Valves) requires system leakage test of Class 1 pressure retaining piping and valves once per 10-year interval conducted at or near the end of each inspection interval. The system leakage test is required to be performed at a test pressure not less than the nominal operating pressure of the RCS corresponding to 100% rated reactor power and shall include all Class 1 components within the RCS boundary.

In Relief Request No. 23, the licensee proposed an alternative to the boundary subject to test pressurization. The line configuration, as outlined, provides double-isolation of the RCS. Under normal plant operating conditions, the subject pipe segments would see RCS temperature and pressure only if leakage through an inboard isolation valve occurs. As requested in Relief Request No. 23, with the inboard isolation valve closed during the system leakage test, the segment of piping between an inboard and an outboard isolation valve would not get pressurized to the required test pressure during a system leakage test. In order to perform the ASME Code-required test, it would be necessary to manually open each inboard isolation valve to pressurize the corresponding pipe segment. Pressurization by this method would preclude double valve isolation of the RCS and the NRC staff concurs that this may cause safety concerns for the personnel performing the examination. The subject isolation valves are located inside the containment, and the NRC staff also concurs that any manual actuation (opening and closing) of these valves would expose plant personnel to undue radiation exposure during modification and restoration of system lineups. Alternatively, the line segments between the isolation valves could be separately pressurized to the required test pressure by a hydrostatic pump but there are no test connections between the isolation valves to attach a pump.

The NRC staff concludes that compliance with the Code requirement would result in hardship without a compensating increase in the level of quality and safety. The licensee has proposed an alternative to visually examine (VT-2) for leaks in the isolated portion of the subject segments of piping with the inboard and outboard isolation valves in the normally closed position which would indicate any evidence of past leakage during the operating cycle as well as any active leakage during the system leakage test if the inboard isolation valve leaks. The staff believes that the licensee's proposed alternative will provide reasonable assurance of structural integrity for the RCS drain lines and the piping segments between an inboard and an outboard isolation valve while maintaining personnel radiation exposure as low as reasonably achievable. Furthermore, the staff notes that there is no known degradation mechanism, such as intergranular stress-corrosion cracking, primary water stress-corrosion cracking, or thermal fatigue that is likely to affect the welds in the subject segments.

5.0 CONCLUSION

The NRC staff concludes that test pressurization during the system leakage test of the Class 1 pressure retaining components within the system boundary of RCS drain lines and piping segments between an inboard and an outboard isolation valve as required by the Code of Record would result in hardship to the licensee without a compensating increase in the level of quality and safety. The staff further concludes that the licensee's proposed alternative in Relief Request 23 provides a reasonable assurance of structural integrity for the subject piping segments.

Therefore, pursuant to 10 CFR 50.55a(a)(3)(ii), the proposed alternative in Relief Request 23 is authorized for the third 10-year ISI interval at Ginna. All other requirements of the ASME Code, Section XI for which relief has not been specifically requested remain applicable, including a third party review by the Authorized Nuclear Inservice Inspector.

Principal Contributor: Jennifer Gall

Date: May 5, 2009

May 5, 2009

Mr. John Carlin
Vice President R.E. Ginna Nuclear Power Plant
R.E. Ginna Nuclear Power Plant, LLC
1503 Lake Road
Ontario, NY 14519-9364

SUBJECT: RELIEF REQUEST NO. 23 RE: FOURTH INTERVAL ISI PROGRAM CATEGORY B-P EXAMS – 10 YEAR CLASS 1 LEAKAGE EXAM - R.E. GINNA NUCLEAR POWER PLANT (TAC NO. ME0456)

Dear Mr. Carlin:

By letter dated January 29, 2009, R.E. Ginna Nuclear Power Plant, LLC (the licensee), submitted Relief Request No. 23 relating to system pressure tests applicable to the R.E. Ginna Nuclear Power Plant (Ginna) for the fourth 10-year inservice inspection (ISI) interval. The relief request pertains to the boundary subject to test pressurization during performance of a system leakage test conducted at or near the end of the inspection interval. In lieu of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code) requirement to include all Class 1 pressure retaining components within the test boundary, the licensee has proposed an alternative to pressurize up to the inboard isolation valve which would exclude a small segment of the Class 1 pressure boundary from attaining test pressure.

The Nuclear Regulatory Commission staff finds that the licensee's proposed alternative provides reasonable assurance of structural integrity and is, therefore, acceptable. Based on the information provided in Relief Request No. 23, the staff concludes in the enclosed safety evaluation that the licensee's compliance to the ISI Code of Record would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety. Therefore, pursuant to Title 10 of the *Code of Federal Regulations,* 50.55a(a)(3)(ii), the staff authorizes the ISI program alternative proposed in Relief Request No. 23 for the fourth 10-year ISI interval for the Ginna Nuclear Power Plant.

Please contact Douglas Pickett at 301-415-1364 or Douglas.Pickett@nrc.gov if you have any questions.

Sincerely,
/RA/
John Boska, Acting Chief
Plant Licensing Branch I-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-244
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