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

SUBJECT: FOURTH INTERVAL INSERVICE INSPECTION PROGRAM RELIEF REQUEST NO. 19 - DEFERRAL OF REACTOR PRESSURE VESSEL CATEGORY B-F EXAMS FROM 2009 TO 2011 - R.E. GINNA NUCLEAR POWER PLANT (TAC NO. MD8733)

Dear Mr. Carlin:

By letter dated May 10, 2008, R.E. Ginna Nuclear Power Plant, LLC, the licensee, submitted Relief Request No. 19 associated with the Fourth 10-Year Interval Inservice Inspection Program for the R.E. Ginna Nuclear Power Plant. Relief Request No. 19 was subsequently resubmitted by letter dated June 23, 2008, as supplemented by letter dated October 31, 2008. The relief request would defer reactor pressure vessel Category B-F Examinations from the 2009 to the 2011 refueling outage and was proposed in accordance with paragraph 50.55a(a)(3)(ii) of Title 10 of the *Code of Federal Regulations* (10 CFR).

The Nuclear Regulatory Commission (NRC) staff concludes that the licensee's proposed alternative provides reasonable assurance of structural integrity and that compliance with the specified requirements of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), Section XI, would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety. Therefore, pursuant to 10 CFR 50.55a(a)(3)(ii), the NRC staff authorizes the extension of the fourth 10-year inservice inspection interval for less than 6 months beyond the 10-year Code inspection interval and the 1-year interval extension provided by IWA-2430(d) for Examination Category B-F welds.

Sincerely,

Mark G. Kowal, 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

February 17, 2009

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

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***/RA/***
Mark G. Kowal, 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

FOURTH 10-YEAR INSERVICE INSPECTION INTERVAL

RELIEF REQUEST NO. 19

R.E. GINNA NUCLEAR POWER PLANT

R.E. GINNA NUCLEAR POWER PLANT, LLC

DOCKET NO. 50-244

## 1.0 INTRODUCTION

By letter dated May 10, 2008 (Reference 1), R.E. Ginna Nuclear Power Plant, LLC, the licensee, requested Nuclear Regulatory Commission (NRC) approval of Relief Request No. 19 that would defer the applicable B-F weld examinations from the 2009 to the 2011 Refueling Outage at the R.E. Ginna Nuclear Power Plant (Ginna Station). Relief Request No. 19 was subsequently resubmitted by letter dated June 23, 2008 (Reference 2), as supplemented by letter dated October 31, 2008 (Reference 3).

The licensee's request is to obtain a less than 6-month interval extension beyond the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), Section XI, allowed 12-month extension (IWA-2430(d)) in order to allow the subject examinations to be performed at the same time as the reactor vessel weld examinations, along with other reactor vessel related examinations. Performing these inspections separate in time from the reactor vessel shell welds would result in hardship without a compensating increase in quality or safety. The fourth 10-year inservice inspection (ISI) interval for Ginna Station started on January 1, 2000, and will end on December 31, 2009.

## 2.0 REGULATORY EVALUATION

Title 10 to the *Code of Federal Regulations* (10 CFR), Section 50.55a(g) specifies that ISI of nuclear power plant components shall be performed in accordance with the requirements of the ASME Code, Section XI, except where specific written relief has been granted by the Commission pursuant to 10 CFR 50.55a(g)(6)(i). 10 CFR 50.55a(g)(6)(i) states that the Commission may grant such relief and may impose such alternative requirements as it determines is authorized by law and will not endanger life or property or the common defense and security and is otherwise in the public interest, given the consideration of the burden upon the licensee. 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 (including supports) shall meet the requirements, except the design and access provisions and preservice examination requirements, set forth in the ASME Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," to the extent practicable 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 therein. The applicable ASME Code, Section XI of record for the fourth 10-year ISI interval at the Ginna facility is the 1995 Edition with the 1996 Addenda.

## 3.0 TECHNICAL EVALUATION

### 3.1 Applicable Code Edition and Addenda

The code of record for the fourth 10-year ISI program at Ginna Station is the ASME Code, Section XI, 1995 Edition with the 1996 addenda. The fourth 10-year ISI interval for Ginna Station began on January 1, 2000 and will end on December 31, 2009.

### 3.2 Components for Which Relief is Requested

| Examination Category | Item No. | Description | Weld ID. |
|---|---|---|---|
| B-F | B5.10 | Pressure Retaining Dissimilar Metal Welds in Vessel Nozzles Made of stainless steel metal | PL-FW-II<br>AC-1003-1<br>PL-FW-VII<br>PL-FW-IV<br>AC-1002-1<br>PL-FW-V |

These examination categories and item numbers are from IWB-2500 and Table IWB-2500-1 of the ASME Code, Section XI.

### 3.3 Applicable Code Requirement

IWB-2412, Inspection Program B, requires volumetric examination of essentially 100% of reactor vessel pressure retaining welds identified in Table IWB-2500-1 once each 10-year interval. IWA-2430(d) allows inspection intervals to be extended by as much as 1 year if this adjustment does not cause successive intervals to be altered by more than 1 year.

### 3.4 Licensee Proposed Alternative and Basis for Use

Pursuant to 10 CFR 50.55a(a)(3)(ii), relief is requested on the basis that performing inspections of six dissimilar metal, Examination Category B-F welds separate in time from the reactor vessel shell welds would result in hardship without a compensating increase in quality or safety. The licensee proposes to perform the subject examinations for the fourth inspection interval one refueling cycle beyond the end of the fourth interval. The fourth inspection interval for Ginna Station started on January 1, 2000, and will end on December 31, 2009. The subject examinations are currently scheduled to be performed during the fall 2009 refueling outage. The inspections are proposed to be performed in the subsequent refueling outage in spring

2011. This inspection date is less than 6 months beyond the 10-year Code inspection interval and the 1-year interval extension provided by IWA-2430(d).

Due to access limitations, past volumetric examinations of the subject pressure retaining dissimilar metal nozzle welds have been performed from the nozzle ID at the same time as the inspection of the reactor vessel shell welds. Performing the inspections of the subject welds separate in time from the reactor vessel shell welds would result in hardship without a compensating increase in quality or safety.

The Ginna Station reactor vessel has six dissimilar metal Examination Category B-F welds from the reactor vessel outlet nozzles to the safe-ends/piping/elbows including two on the reactor vessel outlet nozzles, two on the reactor vessel inlet nozzles, and two on the reactor vessel safety injection nozzles. The welds are stainless steel welds that do not contain any Alloy 82 or 182 weld material. The licensee states that, to date, all known incidents of cracking in the pressurized-water reactor (PWR) fleet in reactor vessel Category B-F welds have been attributed to primary water stress corrosion cracking (PWSCC) in susceptible Alloy 82 and 182 weld materials. There have been no known incidents of cracking in non-alloy 82/182 reactor vessel Category B-F welds. Given this industry history, the licensee states that it is not expected that cracking will occur in these welds at Ginna Station.

In terms of ISI inspection history, the Ginna Station is currently in the Fourth ISI Interval Program. To date, the subject welds have been examined three times. Most recently, the examinations were performed in April 1999 in accordance with the 1986 Edition of the ASME Code, Section XI; and no recordable/reportable indications were found.

The licensee states that the reactor pressure vessel body was designed and constructed to ASME Code, Section III, 1965 Edition. Early codes that were used in the construction of the Ginna Station did not contain requirements to ensure that items be made accessible for future examinations. Due to the limitations of early construction codes, the pressure retaining dissimilar metal welds in the vessel nozzles are not accessible and/or easily accessible from the outside; therefore, the nozzle welds examinations have historically been performed from the vessel interior.

The licensee states that performing a volumetric examination of the pressure retaining dissimilar metal vessel nozzle welds from the ID during the same outage as the reactor vessel shell welds will result in a reduction of man-rem exposure. To access all the nozzles and perform all examinations would require the removal of all fuel and the core barrel from the reactor vessel. The licensee states that an unnecessary risk is created by the removal of the core barrel more than once within an inspection interval to perform associated vessel examinations without a compensating increase to quality or safety. The average dose rate in the general area of the vessel nozzles was 145 mRem per hour during past inspections. The highest measured dose rate during past inspections was 232 mRem per hour with a majority of the workers receiving 130-165 mRem per hour dose rates.

The licensee further states that significant radiation exposure reduction can be realized since the same equipment and personnel used for the volumetric examination of the vessel shell welds from the vessel interior can be used to examine the pressure retaining dissimilar metal nozzle welds from the vessel interior. The volumetric examinations of the reactor pressure vessel pressure retaining dissimilar metal vessel nozzle welds in Table IWB-2500-1, Examination Category B-F, Item Number B5.10 have historically been performed during the same outage as the vessel shell welds at the end of the ISI interval.

The licensee states that there are no known indications in the nozzle-to-safe-end/piping/elbows welds for which the extension is requested. The welds for which the subject examinations are conducted are dissimilar metal stainless steel to low alloy steel welds (Reference 4) which are not susceptible to PWSCC. Absent PWSCC, the most credible mechanism for flaws to initiate and grow in these welds is fatigue due to thermal and mechanical cycling from operational transients. ASME cumulative fatigue usage factors were calculated for these locations using a very conservative design duty cycle where the design duty cycle is the combination of the transient characteristics (pressure and temperature with time) and the number of design cycles.

The licensee states that the calculated fatigue usage factors are much less than the ASME Code design limit of 1.0 after 40 years of operation, and typically less than 0.1 in the region. Further, as Ginna Station enters the extended license period, these calculated fatigue usage factors will not exceed 1.0 after 60 years of operation since the originally specified number of cycles for 40 years of operation will now be used for the 60-year life. For this reason, it is very unlikely that a flaw would have initiated during the 10 years since the last ISI. Given the very small number of transients from the design duty cycle that may occur over the period of the requested extension, it is even more unlikely that any flaws will initiate during the requested extension.

The licensee states that in the unlikely event that a flaw was either missed in the previous ISIs or a flaw was initiated since the last inspection, the growth of any existing flaw is expected to be very small over the life of the reactor vessel. For example, flaw evaluation handbooks have been developed and submitted to the NRC for Westinghouse 2 loop plants (References 5 and 6) that have comparable geometries and loading conditions to that of the Ginna Station. The licensee further states that these evaluations, which take into consideration a very conservative design duty cycle, show that even if a surface flaw with an aspect ratio of 6 (l/a) and initial depth of 20% through wall (a/t) is assumed to exist in any of the subject welds, the flaw will remain acceptable for at least 20 years per the ASME Code, Section XI.

### 3.5 Duration of Relief

The alternative is requested to extend the Ginna Station Fourth Inservice Inspection Interval by less than 6 months beyond the ASME Code required 10-year inspection interval and Code allowed 12-month extension such that the end date of Relief Request Number 19 would be May 30, 2011. The Ginna Station Fourth Interval ISI Program will end on December 31, 2009. The ASME Code allowed extension would be in effect until December 31, 2010, for ASME Category and Items Numbers identified in Relief Request Number 19.

## 4.0 STAFF EVALUATION

The licensee proposes to perform the subject examinations for the fourth inspection interval one refueling cycle beyond the end of the fourth interval. Thus, the licensee is seeking relief from the limitation to extend an interval for 1 year and requesting to extend the interval for less than 6 months beyond the ASME Code required 10-year inspection interval and Code allowed 12-month extension. Subarticle IWA-2432, Inspection Program B, states the inspection intervals shall comply with the following, except as modified by subarticle IWA-2430(d) which requires that the fourth inspection interval be the 10 years following the third inspection interval. Subarticle IWA-2430(d) states for components inspected under Program B, each of the inspection intervals may be extended or reduced by as much as 1 year. Adjustments shall not cause successive intervals to be altered by more than 1 year from the original pattern of intervals.

As stated above, the Code states that the 1-year extension will not alter the successive intervals by more than 1 year from their original pattern of intervals. This aspect of the Code is important for components later in service life to assure any adverse trends are discovered. In this case, the licensee plans to maintain the duration of the fourth interval to end 10 years after the end of the third interval for all components except for those ASME Category and Item number that are identified within this Relief Request. The Ginna Station Fifth Interval ISI Program will start on January 1, 2010, and end on December 31, 2019. In Reference 3, the licensee acknowledges that they are required to either perform the examinations again on applicable welds in accordance with the Fifth Interval ISI Program or to generate a new relief request. The licensee's plans are in compliance with Code requirements under subarticle IWA-2430(d) and provide reasonable assurance that any adverse trends will be discovered.

This relief request pertains only to the volumetric examination requirements for the six dissimilar metal Examination Category B-F nozzle-to-safe end, elbow or pipe welds. Normally, these examinations are performed at the same time as the reactor pressure vessel weld examinations from the inside of the vessel as this method of examination provides greater volumetric coverage as compared to performing the examinations from the outside surface. The licensee discussed the past volumetric examinations of the pressure retaining dissimilar metal nozzle welds that have been performed from the nozzle inside diameter (ID) at the same time as the inspection of the reactor vessel shell welds. In Reference 3, the licensee discussed the results of the most recent inspections performed in 1999 from the nozzle ID. All six nozzle weld inspections resulted in an achieved coverage of 100% with no recordable indications. If the welds were to be inspected from the nozzle outside diameter (OD), two of the welds would have 0% coverage and the remaining four would have 50% coverage due to physical obstructions. The NRC staff agrees with the licensee that inspection from the ID provides a greater volumetric coverage (100% for these six welds). Thus, the examination from the ID is more likely to identify any significant flaws that may challenge the structural integrity of the welds.

The Ginna Station reactor vessel has six dissimilar metal Examination Category B-F welds from the reactor vessel outlet nozzles to the safe-ends/piping/elbows including two on the reactor vessel outlet nozzles, two on the reactor vessel inlet nozzles, and two on the reactor vessel safety injection nozzles. The welds are stainless steel welds that do not contain any Alloy 82 or 182 weld material. No indications have been recorded in the course of ISI inspections of these welds at Ginna Station. In terms of industry experience, the licensee states that, to date, all known incidents of cracking in the PWR fleet in reactor vessel Category B-F welds have been attributed to PWSCC in susceptible Alloy 82 and 182 weld materials. There have been no known incidents of cracking in non-alloy 82/182 reactor vessel Category B-F welds. Furthermore, the licensee notes in Reference 3 that, while indications have been recorded in the course of performing ISI inspections at other plants, these indications have not been determined to be structurally significant and are most often smaller than the acceptance standards of the ASME Code, Section XI, IWB-3500. These indications are attributed to fabrication, rather than service induced flaws, since they were typically embedded flaws. Given this industry history, the licensee states that it is not expected that cracking will occur in these welds at Ginna Station. The staff concludes that the Ginna Station ISI results, coupled with the PWR fleet ISI history, would indicate that service induced cracking in these welds is not expected.

The licensee states that performing a volumetric examination of the pressure retaining dissimilar metal vessel nozzle welds from the ID during the same outage as the reactor vessel shell welds will result in a reduction of man-rem exposure. To access all the nozzles and perform all examinations requires the removal of all fuel and the core barrel from the reactor vessel.

Deferring the examinations until the performance of the complete reactor vessel ISI will consolidate activities and reduce personnel radiological exposure as the same equipment and personnel used for the volumetric examination of the reactor vessel shell welds from the vessel interior can be used to examine the pressure retaining dissimilar metal nozzle welds from the vessel interior. The staff agrees with the licensee that performing the examinations in two separate outages creates additional man-rem exposure which can be avoided by performing all of the inspections during one outage.

The licensee states that the welds for which the subject examinations are conducted are dissimilar metal stainless steel to low alloy steel welds (Reference 4) which are not susceptible to PWSCC. The licensee also states that absent PWSCC, the most credible mechanism for flaws to initiate and grow in these welds is fatigue due to thermal and mechanical cycling from operational transients. ASME cumulative fatigue usage factors were calculated for these weld locations and the results were much less than the ASME Code design limit of 1.0 after 40 years of operation, and typically less than 0.1 in the region. Furthermore, the licensee states that, as Ginna Station enters the extended license period, these calculated fatigue usage factors will not exceed 1.0 after 60 years of operation since the originally specified number of cycles for 40 years of operation bounds the cycle for the 60-year life. For this reason, the licensee contends that it is very unlikely that a flaw would have initiated during the 10 years since the last ISI and, that given the very small number of transients from the design duty cycle that may occur over the period of the requested extension, it is even more unlikely that any flaws will initiate during the requested extension. The NRC staff agrees these stainless steel welds are not susceptible to PWSCC, and that absent that flaw initiation mechanism, fatigue due to thermal and mechanical cycling is the most credible mechanism for flaw initiation. Additionally, the staff agrees that due to the small number of transients that may occur during the requested extension, it is unlikely that any flaws will initiate during the requested extension.

The licensee states that in the unlikely event that a flaw was either missed in the previous ISIs or a flaw had initiated since the last inspection, the growth of any existing flaw is expected to be very small over the life of the reactor vessel. The licensee cites flaw evaluation handbooks that were developed and submitted to the NRC for two other Westinghouse 2 loop plants that have comparable geometries and loading conditions to that of Ginna Station. These evaluations, which took into consideration a very conservative design duty cycle, show that even if a surface flaw with an aspect ratio of 6 (l/a) and initial depth of 20% through wall (a/t) is assumed to exist in any of the subject welds, the flaw will remain acceptable for at least 20 years per the ASME Code, Section XI. In its request for additional information (RAI), the NRC staff asked the licensee to provide a detailed comparison of piping geometries and loading conditions to demonstrate that the flaw evaluations for the two other Westinghouse plants were applicable to Ginna Station. In their RAI response (Reference 3), the licensee provided detailed comparisons of the dimensions, design basis transients, and nozzle loads for their plant as well as the other two plants. The NRC staff notes that the Ginna Station dimensions, design basis transients and nozzle loads are identical or nearly identical to those of the other two plants. Therefore, the NRC staff has determined that the licensee has demonstrated that the flaw evaluation handbooks for these other plants are applicable to Ginna Station. As such, the staff agrees that in the unlikely event that a flaw was either missed in the previous ISI inspection or one had initiated since the last inspection, the flaw growth would be very small over the life of the reactor vessel and would remain acceptable for a time period well exceeding the length of time of the requested extension.

Lastly, in its RAI, the NRC staff asked the licensee to describe what leakage detections systems and system walkdowns are in place to detect leakage in these welds. In Reference 3, the

licensee explained that an ASME Code, Section XI, Class 1 system leakage examination will be performed with qualified VT-2 examiners at normal operating pressure and temperature following the 2009 refueling outage. Additionally, radiation monitoring instrumentation is available to detect any reactor coolant that may be released to containment. The staff finds these additional measures provide a high level of confidence that any leakage from these welded joints would be promptly identified.

In summary, the NRC staff believes that conducting the examinations required for Examination Category B-F dissimilar metal welds in the same outage as the examinations of the reactor shell welds will consolidate activities and reduce personnel radiological exposure. Examination of the subject items during the upcoming refueling outage (fall 2009) would not provide an additional level of safety or quality in comparison to deferring the examinations for one refueling cycle to the spring 2011 outage. The staff finds the hazard associated with the one-cycle extension of the examination interval is sufficiently small and coupled with the dose saving that would result, the alternative provides reasonable assurance of structural integrity and that compliance with the specified requirements of ASME Code, Section XI would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

## 5.0 CONCLUSION

Based on the above evaluation, the NRC staff concludes that the licensee's proposed alternative provides reasonable assurance of structural integrity and that compliance with the specified requirements of ASME Code, Section XI would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety. Therefore, pursuant to 10 CFR 50.55a(a)(3)(ii), the NRC staff authorizes the extension of the fourth 10-year ISI interval for less than 6 months beyond the 10-year Code inspection interval and the 1-year interval extension provided by IWA-2430(d) for Examination Category B-F welds.

All other requirements of the ASME Code for which relief has not been specifically requested remain applicable including third party review by the Authorized Nuclear Inservice Inspector.

## 6.0 REFERENCES

1. Letter from J. Pacher, Constellation Energy, to NRC dated May 10, 2008, transmitting "Fourth Ten-Year Interval Inservice Inspection Program Submittal of Relief Request Numbers 18, 19, 20, and 21" (Agencywide Documents Access and Management System (ADAMS) Accession No. ML081400722).

2. Letter from J. Pacher, Constellation Energy, to NRC dated June 23, 2008, transmitting "Fourth Ten-Year Interval Inservice Inspection Program Withdrawel of Relief Request Number 18 and Re-Submittal of Relief Request Number 19" (ADAMS Accession No. ML081790151).

3. Letter from J. Pacher, Constellation Energy, to NRC dated October 31, 2008, transmitting "Responses to Request for Additional Information Pertaining to Fourth 10-Year Inservice Inspection Program Relief Request Number 19" (ADAMS Accession No. ML083110415).

4. E-mail from D. Pickett, NRC, to T. Harding, Jr., Constellation Energy, dated December 18, 2008, "Clarification to Application for Ginna Relief Request No. 19" (ADAMS Accession No. ML083530072).

5. WCAP-10363, “Handbook on Flaw Evaluation for Prairie Island Units 1 & 2 Reactor Vessels,” December, 1984.

6. WCAP-11477, Revision 1, “Handbook on Flaw Evaluation for Point Beach Units 1 & 2 Reactor Vessels,” July 1990.

Principal Contributor: Carol Nove

Date: