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Nuclear Generation Group

June 23, 2008

U. S. Nuclear Regulatory Commission
Washington, DC 20555

ATTENTION: Document Control Desk

SUBJECT: **R.E. Ginna Nuclear Power Plant**
Docket No. 50-244

Fourth Ten-Year Interval Inservice Inspection Program
Withdrawal of Relief Request Number 18 and
Re-submittal of Relief Request Number 19

Reference: (1) Letter from J. Pacher, Ginna LLC, to NRC Document Control Desk, Subject: Fourth Ten-Year Interval Inservice Inspection Program Submittal of Relief Request Numbers 18, 19, 20, and 21, dated May 10, 2008.

(2) Letter from D. Pickett, NRC, to J. Carlin, Ginna LLC, Subject: R.E. Ginna Nuclear Power Plant – Acceptance Review Regarding Fourth Ten-Year Interval Inservice Inspection Program Submittal of Relief Request Numbers 18, 19, 20, and 21 (TAC NOS. MD8732 – MD8735), dated June 18, 2008.

In Reference 1, R.E. Ginna Nuclear Power Plant, LLC (Ginna LLC) submitted four proposed code relief requests associated with the Fourth Ten-Year Interval Inservice Inspection Program. Based on conversations with the NRC staff on June 11, 2008 (as documented in Reference 2) it has been determined that due to the recent NRC approval of WCAP-16168, which was the technical basis for our submittal, Ginna LLC is requesting withdrawal of our ISI-18 code relief request attachment from Reference 1. We will formally request the 20 year frequency for the subject inspections in a future submittal (in approximately six weeks).

Also as the result of the June 11, 2008 conversation, Ginna LLC is submitting a replacement proposed code relief request for our ISI-19 attachment in Reference 1. The attached proposed code relief revises the basis for the request and removes the risk related information.

No new commitments are being made in this letter.

Should you have questions regarding this matter, please contact Tom Harding (585) 771-3384, or Thomas.harding@constellation.com.

Very truly yours,

Joseph E. Pacher

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NRR

1001973

Attachment: RELIEF REQUEST NO. 19

cc: S. J. Collins, NRC
D. V. Pickett, NRC
Ginna Resident Inspector, NRC

RELIEF REQUEST NO. 19
R.E. Ginna Nuclear Power Plant – Fourth Interval ISI Program
Defer RPV Category B-F Exams from 2009 to 2011 Outage

RELIEF REQUEST NO. 19
R.E. Ginna Nuclear Power Plant – Fourth Interval ISI Program
Defer RPV Category B-F Exams from 2009 to 2011 Outage

Proposed Alternative
In Accordance with 10 CFR 50.55a(a)(3)(ii)
-Compliance with the Specified Requirements Would Result in Hardship or Unusual Difficulty without a Compensating Increase in the Level of Quality and Safety-

1. ASME Code Component(s) Affected

The affected components are the R.E. Ginna reactor vessel nozzle-to-safe-end/piping welds, specifically the following ASME Boiler and Pressure Vessel (BPV) Code Section XI (Reference 1) Examination Category and Item Number. This examination category and item number is from IWB-2500 and Table IWB-2500-1 of the ASME BPV Code Section XI.

Examination Category	Item No.	Description
B-F	B5.10	Pressure Retaining Dissimilar Metal Welds in Vessel Nozzles

(Throughout this request the above examination category is referred to as "the subject examinations" and the ASME BPV Code Section XI is referred to as "the Code". "Inspections" and "Examinations" may be used interchangeably.)

2. Applicable Code Edition and Addenda

The R.E. Ginna Fourth Interval Inservice Inspection (ISI) Program Plan is prepared to the ASME Section XI Code, 1995 Edition with the 1996 Addenda.

3. Applicable Code Requirement

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

4. Reason for Request

Relief Request Number 19 is being submitted along with Relief Request Numbers 18, 20 and 21. All four relief requests are intended to address deferral of the associated Reactor Pressure Vessel related examinations from the 2009 Refueling Outage to the 2011 Refueling Outage. This request is to obtain a less than six month interval extension beyond the Code 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 (Relief Request Number 18), along with other reactor vessel related examinations.

5. Proposed Alternative and Basis for Use

R.E. Ginna Nuclear Power Plant 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 R. E. Ginna 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 ten year Code inspection interval and the one year interval extension provided by IWA-2430(d).

Due to access limitations, past volumetric examinations of the 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 these inspections separate in time from the reactor vessel shell welds would, in accordance with 10 CFR 50.55a (a)(3)(ii), result in hardship without a compensating increase in quality or safety.

The technical justification for the deferral of the subject examinations consists of three areas. These are:

A.) PWR Service Experience
B.) R.E. Ginna Inservice Inspection History, Access Limitations and Radiation Exposure Reduction
C.) Deterministic Flaw Growth Analysis

A.) PWR Service Experience

The Ginna Station reactor vessel has six (6) dissimilar metal Examination Category B-F welds from the reactor vessel nozzles to the safe-ends/piping/elbows. These welds exist on the reactor vessel outlet nozzles (2), reactor vessel inlet nozzles (2), and the reactor vessel safety injection nozzles (2). These welds are stainless steel welds that do not contain any Alloy 82 or 182 weld material. To date, all known incidents of cracking in the 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 history, it is not expected that cracking will occur in these welds at Ginna Station.

B.) R.E. Ginna Inservice Inspection History, Access Limitations and Radiation Exposure Reduction

R. E. Ginna Nuclear Power Plant is currently in the Fourth Interval ISI Program. The subject examinations have been performed three times for Inservice Inspection. Most recently, these examinations were performed in April of 1999 in accordance with the 1986 Edition of the ASME Boiler and Pressure Vessel Code, Section XI. Table 1 provides a summary of the inservice inspection results from the last inspection. Due to the improvements in inspection technology with time, the most recent inspection is considered to be of the greatest quality of the three inservice inspections performed. No indications were identified as reflected by Table 1 below.

Table 1: R.E. Ginna Examination Category B-F Inservice Inspection Results

Weld ID	Description	# of recordable indications	# of reportable indications
PL-FW-II	30° Outlet Nozzle-to-safe-end from nozzle and safe-end side (ISI Summary # I 002100)	0	0
AC-1003-1	105° SI Nozzle-to-safe-end from nozzle and safe-end side (ISI Summary # I 003300)	0	0
PL-FW-VII	150° Inlet Nozzle-to-safe-end from nozzle and safe-end side (ISI Summary # I 003000)	0	0
PL-FW-IV	210° Outlet Nozzle-to-safe-end from nozzle and safe-end side (ISI Summary # I 002700)	0	0
AC-1002-1	285° SI Nozzle-to-safe-end from nozzle and safe-end side (ISI Summary # I 003600)	0	0
PL-FW-V	330° Inlet Nozzle-to-safe-end from nozzle and safe-end side (ISI Summary # I 002400)	0	0

The reactor pressure vessel body was designed and constructed to ASME Section III, 1965 Edition. Early Codes that were used in the construction of 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 vessel nozzles are not accessible and/or easily accessible from the outside. The nozzle welds have historically been performed from the vessel interior.

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. 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.

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 inservice inspection interval.

C.) Deterministic Flaw Growth Analysis

As shown in the inservice inspection results in paragraph "B", 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 similar metal low alloy steel welds 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 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 R.E. Ginna 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 inservice inspection. 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.

In the unlikely event that a flaw was either missed in the previous inservice inspections (discussed in paragraph "B") 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 2 and 3) that have comparable geometries and loading conditions to that of

R.E. Ginna. 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.

6. Duration of Proposed Alternative

The alternative is requested to extend the Fourth Inservice Inspection Interval by less than 6 months beyond the ASME Code required 10-year inspection interval and Code allowed twelve month extension. This request is applicable to R.E. Ginna fourth inspection interval only.

7. References

1. *ASME Boiler and Pressure Vessel Code,* Section XI, 1995 Edition with the 1996 Addenda, American Society of Mechanical Engineers, New York.
2. WCAP-10363, "Handbook on Flaw Evaluation for Prairie Island Units 1 & 2 Reactor Vessels," December 1984.
3. WCAP-11477, Revision 1, "Handbook on Flaw Evaluation for Point Beach Units 1 & 2 Reactor Vessels," July 1990.