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Constellation Energy
Nuclear Generation Group

October 3, 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
Re-submittal of Relief Request Number 18

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.

(3) Letter from J. Pacher, Ginna LLC, to NRC Document Control Desk, Subject: Fourth Ten-Year Interval Inservice Inspection Program Withdrawal of Relief Request Number 18 and Re-submittal of Relief Request Number 19, dated June 23, 2008.

(4) Letter from Ho K. Nieh, NRC, to Gordon Bischoff, WOG, regarding Final Safety Evaluation for PWROG Topical Report WCAP-16168-NP, Revision 2, (TAC NO. MC9768), dated May 8, 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Based on conversations with the NRC staff on June 11, 2008 (as documented in Reference 2), Relief Request Number 18 was withdrawn (Reference 3). This letter provides the re-submittal of that relief request.

The NRC approved WCAP-16168-NP-A, Revision 2, "Risk-Informed Extension of The Reactor Vessel In-Service Inspection Interval," in Reference 4. This WCAP provides for extension of the inservice inspection interval for certain pressure retaining welds in the reactor vessel from 10 to 20 years. Ginna LLC proposes to implement this extended inservice inspection interval for the R.E. Ginna Nuclear Power Plant. The plant specific information identified by Reference 4 as needed to support this request is provided in Attachment 1. Ginna LLC has concluded that the proposed alternative provides an acceptable level of quality and safety. The relief is requested under the provisions of 10CFR 50.55a(a)(3)(i).

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As required by Reference 4, and under separate cover, Ginna LLC is requesting an amendment to the R.E. Ginna license that will require that the information and analyses requested in the final rule for 10 CFR 50.61a, Section (e) or, prior to issuance, the proposed rule (72 FR 56275) for 10 CFR 50.61 a, Section (e) be submitted within one year of completing each of the ASME Code, Section XI, Category B-A and B-D reactor vessel weld inspections.

No new commitments are being made in this letter.

Ginna LLC requests NRC approval of the attached relief request by April 3, 2009 to support planning for the fall 2009 refueling outage.

Should you have questions regarding this matter, please contact David F. Wilson (585) 771-5219.

Very truly yours,

Joseph E. Pacher

Attachment: RELIEF REQUEST NO. 18

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

RELIEF REQUEST NO. 18
R.E. Ginna Nuclear Power Plant – Fourth Interval ISI Program
Reactor Vessel Inservice Inspection Interval Extension

RELIEF REQUEST NO. 18
R.E. Ginna Nuclear Power Plant – Fourth Interval ISI Program
Reactor Vessel Inservice Inspection Interval Extension

Proposed Alternative
In Accordance with 10 CFR 50.55a(a)(3)(i)
-Alternative Provides Acceptable Level of Quality and Safety-

1. ASME Code Component(s) Affected

The affected component is the R.E. Ginna reactor vessel, specifically the following American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (BPV) Code Section XI (Reference 1) examination categories and item numbers covering examinations of the reactor vessel (RV). These examination categories and item numbers are from IWB-2500 and Table IWB-2500-1 of the ASME BPV, Code Section XI.

Examination Category	Item No.	Description
B-A	B1.11	Circumferential Shell Welds
B-A	B1.30	Shell-to-Flange Weld
B-D	B3.90	Nozzle-to-Vessel Welds
B-D	B3.100	Nozzle Inner Radius Areas

(Throughout this request the above examination categories are referred to as "the subject examinations" and the ASME BPV Code, Section XI, is referred to as "the Code.")

2. Applicable Code Edition and Addenda

ASME Code Section XI, "Rules and Inservice Inspection of Nuclear Power Plant Components," 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. The R. E. Ginna fourth inspection interval is scheduled to end on or before December 31, 2009.

4. Reason for Request

Relief Request Number 18, which is being submitted along with Relief Request Numbers 19, 20 and 21, deals with the Reactor Vessel Inservice Inspection Interval extension to allow the subject examinations to be performed at the same time along with other reactor vessel related examinations during the 2011 outage and the next exam, if applicable, to be performed in 2031.

An alternative is requested from the requirements of IWA-2412, Inspection Program B, that volumetric examination of essentially 100% of reactor pressure vessel retaining welds, Examination Categories B-A and B-D welds, be performed once each ten-year interval. Extension of the inspection interval for Examination Category B-A and B-D welds from 10 years to up to 20 years will result in a reduction in personnel radiation exposure and examination costs.

5. Proposed Alternative and Basis for Use

R.E. Ginna Nuclear Power Plant proposes to defer the ASME Code required volumetric examination of the Reactor Pressure Vessel full penetration retaining Category B-A and B-D welds for the fourth inservice inspection until 2011 and to perform the inservice inspection on a twenty-year inspection interval, instead of the currently required ten-year inspection interval. Therefore, the fifth inservice inspection is proposed to be performed in 2031. These dates are consistent with the information provided to the Staff in PWR Owners Group letter OG-06-356 (Reference 2).

In accordance with 10 CFR 50.55a(a)(3)(i), the proposed alternate inspection interval, which provides an acceptable level of quality and safety, is requested on the basis that the current inspection interval can be extended based on a negligible change in risk by satisfying the risk criteria specified in Regulatory Guide 1.174 (Reference 3).

The methodology used to demonstrate the acceptability of extending the fourth and fifth inspection intervals for Category B-A and B-D welds based on a negligible change in risk is contained in the latest revision of WCAP-16168-NP-A (Reference 4). This methodology was used to develop a pilot plant analysis for Westinghouse, Combustion Engineering, and Babcock and Wilcox reactor vessel designs and is an extension of the work that was performed as part of the Nuclear Regulatory Commission Pressurized Thermal Shock (PTS) Risk Re-Evaluation (NUREG-1874, Reference 5). The critical parameters for demonstrating that this pilot plant analysis is applicable on a plant specific basis, as identified in the latest revision of WCAP-16168-NP, are identified in Table 1. By demonstrating that each plant specific parameter is bounded by the corresponding pilot plant parameter, the application of the methodology to the R.E. Ginna reactor vessel is acceptable as shown in Table 1 below.

Table 1 Critical Parameters for Application of Bounding Analysis

Parameter	Pilot Plant Basis	Plant Specific Basis	Additional Evaluation Required?
Dominant Pressurized Thermal Shock (PTS) Transients in the NRC PTS Risk Study are applicable	NRC PTS Risk Study (Reference 5)	Ginna is bounded by PTS Generalization Study (Reference 6)	No
Through Wall Cracking Frequency	1.76E-08 Events per year (Reference 4)	7.47E-12 Events per year (Calculated per Reference 5)	No
Frequency and Severity of Design Basis Transients	7 heatup/cooldowns per year (Reference 4)	Ginna is bounded by 7 heatup/cooldowns per year	No
Cladding Layers (Single/Multiple)	Single Layer (Reference 4)	Single Layer	No

Additional information relative to R.E. Ginna reactor vessel inspections is provided in Table 2. This information confirms that satisfactory examinations have been performed on the R.E. Ginna reactor vessel.

Table 2 Additional Information Pertaining to Reactor Vessel Inspection	
Inspection methodology:	Category B-A welds: ASME Section XI Appendix VIII (Reference 1) Category B-D welds: Regulatory Guide 1.150 (Reference 7) Information is for most recent inservice inspection performed in 1999.
Number of past inspections:	A minimum of 3 inspections have been performed to date on each Category B-A and B-D weld.
Number of indications found:	Zero reportable indications have been found to date. Any recordable indications have been acceptable per ASME Section XI IWB-3500. One recordable indication is in the inner 1/8th of the vessel inside diameter in the beltline region. The indication has a depth of 0.22", length of 1.54" and is 0.38" subsurface. This indication is acceptable per the flaw acceptance criteria in the proposed voluntary PTS Rule, 10 CFR 50.61a (Reference 8).
Proposed inspection schedule for balance of plant life:	Reactor Pressure Vessel Examination Category B-A and B-D inspections are currently scheduled for the 2009 outage within the Fourth Interval Inservice Inspection Program. These examinations for the Fourth Interval Inservice Inspection Program are proposed to be performed in the 2011 outage. The following examinations, if applicable, are proposed to be performed in 2031.

The information in Table 3 is identified in WCAP-16168-NP-A, Revision 2, as additional information to be provided relative to the Through-Wall Cracking Frequency (TWCF) calculation for R.E. Ginna Nuclear Power Plant.

Table 3 Details of Through-Wall Cracking Frequency (TWCF) Calculation

Inputs

Reactor Coolant System Temperature, T_{RCS}[°F]: 544.2 | T_{wall} (inches): 6.50

#	Region/Component Description	Material	Cu [wt%]	Ni [wt%]	P [wt%]	Mn [wt%]	Un-Irradiated $RT_{NDT(u)}$ [°F]	Fluence [10^{19} Neutron/cm^2, E>1 MeV]
1	Inter./Lower Circ. Weld	Linde 80	.250	.560	.012	1.35	-4.8	5.42
2	Nozzle/Inter. Circ Weld	Linde 80	.230	.590	.021	1.35	10.0	0.516
3	Nozzle Forging	A508-2	.070	.680	.010	.700	30.0	0.516
4	Lower forging	A508-2	.050	.690	.010	.700	40.0	5.42
5	Intermediate Forging	A508-2	.070	.690	.010	.700	20.0	5.42

Outputs

Methodology Used to Calculate ΔT_{30}: NUREG-1874

	Region # (From Above)	RT_{MAX-XX} [R]	Fluence [10^{19} Neutron/cm^2, E>1 MeV]	ϕ (flux) [Neutrons/ cm^2/sec]	ΔT_{30} [°F]	$TWCF_{95-XX}$
Forging - FO	4	562.00	5.42	2.86E+10	62.31	7.63E-14
Circumferential Weld - CW	1	684.80	5.42	2.86E+10	229.91	3.40E-12
$TWCF_{95-TOTAL}$ ($\alpha_{AW}TWCF_{95-AW} + \alpha_{PL}TWCF_{95-PL} + \alpha_{CW}TWCF_{95-CW} + \alpha_{FO}TWCF_{95-FO}$):						7.47E-12

The methodology used to demonstrate the acceptability of extending the Reactor Pressure Vessel, Category B-A and B-D welds, is based on a negligible change in risk as contained in the latest revision of WCAP-16168-NP-A (Reference 4). By demonstrating that each plant specific parameter is bounded by the corresponding pilot plant parameter, the application of the methodology to the R.E. Ginna reactor vessel is acceptable as shown in the Tables above. WCAP-16168-NP-A was developed to decrease the frequency of examinations from once every ten years to once every twenty years.

6. Duration of Proposed Alternative

This request is applicable to the R. E. Ginna Inservice Inspection Program for the remainder of the renewed licensed lifetime of the Reactor Pressure Vessel for the examination categories B-A and B-D welds.

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. F. P. Schiffley to U. S. NRC, PWR Owners' Group Letter, OG-06-356, "Plan for Plant Specific Implementation of Extended Inservice Inspection Interval per WCAP-16168-NP, Revision 1, "Risk-Informed Extension of the Reactor Vessel In-Service Inspection Interval." MUHP 5097-99, Task 2059, "October 31, 2006.
3. NRC Regulatory Guide 1.174, Revision 1, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," November 2002.
4. WCAP-16168-NP-A, Revision 2, "Risk-Informed Extension of the Reactor Vessel In-Service Inspection Interval," June 2008
5. NUREG-1874, "Recommended Screening Limits for Pressurized Thermal Shock," March, 2007.
6. NRC Letter Report, "Generalization of Plant-Specific Pressurized Thermal Shock (PTS) Risk Results to Additional Plants," December 14, 2004.
7. NRC Regulatory Guide 1.150, Revision 1, "Ultrasonic Testing of Reactor Vessel Welds During Preservice and Inservice Examinations," February 1983.
8. SECY-07-0104, "Proposed Rulemaking - Alternate Fracture Toughness Requirements for Protection against Pressurized Thermal Shock," June 25, 2007, Enclosure 1.