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November 21, 2002

U.S. Nuclear Regulatory Commission
Document Control Desk
Attn: Mr. Russell Arrighi (Mail Stop O-12D-3)
Office of Nuclear Reactor Regulation
Washington, D.C. 20555-0001

Subject: Topical Reports used in License Renewal Application for Time Limited Aging Analyses
R. E. Ginna Nuclear Power Plant
Docket No. 50-244

Dear Mr. Arrighi:

In the Ginna Station License Renewal Application (LRA), submitted 7/30/2002, several Time-Limited Aging Analyses (TLAAs) were described in Section 4.0. RG&E is providing the following information to demonstrate that the topical reports used as the basis for our conclusions all used methodologies that were previously reviewed and approved by the NRC or are currently already under NRC review. These references should therefore not be required to be submitted for NRC review and approval by RG&E in order to process the Ginna LRA. They are available onsite for inspection

Section 4.2.1 Upper Shelf Energy

The topical report which provides the basis for the Ginna reactor vessel equivalent margins analysis is BAW-2425, Rev. 1, "Low Upper Shelf Toughness Fracture Mechanics Analysis of Reactor Vessel of R.E. Ginna for Extended Life of 54 Effective Full Power Years", May 2002.

The same methodology is employed as was previously reviewed and approved in topical reports BAW-2192-PA, "Low Upper Shelf Fracture Mechanics Analysis of Reactor Vessels of B&W Reactor Vessel Working Group for Level A and B Service Loads", April 1994, and BAW-2178-PA, "Low Upper-Shelf Fracture Mechanics Analysis of Reactor Vessels of B&W Reactor Vessel

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Working Group for Level C and D Service Loads, "April 1994. The NRC Safety Evaluation Reports approving these topical reports are contained within the reports, and are both dated March 29, 1994.

Section 4.2.2 Pressurized Thermal Shock

This TLAA was performed to show conformance to 10 CFR 50.61, using BAW-1803, Rev. 1 "Correlations for Predicting the Effects of Neutron Radiation on Linde-80 Submerged Arc Welds" as a reference. Reactor vessel data was collected in accordance with the requests in Generic Letter 92-01, Rev. 1, Supp. 1, "Reactor Vessel Integrity". Fluence calculation methodology was consistent with Regulatory Guide 1.190. Credible surveillance data was available, but more conservative generic data and irradiation shift calculations in accordance with Regulatory Guide 1.99, Rev. 2 were used. The NRC SER dated 7/6/99 accepted the RG&E responses to Generic Letter 92-01, Rev. 1, Supp. 1, and this information was incorporated into the NRC's "Reactor Vessel Integrity Database", RVID, Version 2.

Section 4.2.3 Pressure-Temperature Limits

WCAP-15885, "R.E. Ginna Heatup and Cooldown Limit Curves for Normal Operations", May 2002 was developed to establish these limits. Regulatory Guides 1.190 and 1.99 Rev. 2 were used to establish fluence and ΔRT_{ndt} values. ASME Code Case N-641, "Alternative Pressure-Temperature Relationship and Low Temperature Overpressure Protection System Requirements Section XI, Division 1", January 17, 2000 is the primary reference for the methodology employed in WCAP-15885. ASME Code Cases N-640, "Alternative Reference Fracture Toughness for Development of P-T Limit Curves for Section XI, Division 1", February 26, 1999, and N-588, "Attenuation to Reference Flaw Orientation of Appendix G for Circumferential Welds in Reactor Vessels, Section XI, Division 1", December 12, 1997 were also employed.

These code cases have been published in Draft Regulatory Guide DG-1091 (proposed Revision 13 to RG1.147), "Inservice Inspection Code Case Acceptability, ASME Section XI, Division 1", December 2001. Since they are already in the NRC review and approval process, they should not require submittal on the Ginna Station docket.

Section 4.7.7 Thermal Aging of Cast Austenitic Stainless Steel

WCAP 15837, "Technical Justification for Eliminating Large Primary Loop Pipe Ruptures as the Structural Design Basis for the R.E. Ginna Nuclear Power Plant for the License Renewal Program", April 2002 contains a fracture mechanics analysis for the RCS cast austenitic stainless steel elbows. The methodology employed in this analysis is consistent with NUREG/CR 4513 Rev. 1, which is referenced as an acceptable means of determining fracture toughness in NUREG-1801, GALL program XI.M12.

WCAP-15873, "A Demonstration of the Applicability of ASME Code Case N-481 to the Primary Loop Casings of R.E. Ginna Nuclear Power Plant for the License Renewal Programs", April 2002 contains a fracture mechanics analysis of the cast austenitic stainless steel reactor coolant pump casings. Code Case N-481 has been reviewed by the NRC, and approved for use by Regulatory Guide 1.147, "Inservice Inspection Code Case Acceptability ASME Section XI Division 1".

Very truly yours,

Robert C. Mecredy

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