

July 30, 2003

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: RAI Response Clarifications and Commitment Schedule Update
R. E. Ginna Nuclear Power Plant
Docket No. 50-244

Dear Mr. Arrighi:

Attachment 1 to this letter provides responses to clarifications resulting from NRC review of our previous RAI responses. Attachment 2 provides a completion schedule for commitments provided with our License Renewal Application of July 30, 2002.

I declare under penalty of perjury under the laws of the United States of America that I am authorized by RG&E to make this submittal and that the foregoing is true and correct.

Very truly yours,

Executed on July 30, 2003


Robert C. Mecredy

Attachments

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List of Regulatory Commitments

The following table identifies those actions committed to by Rochester Gas & Electric (RG&E) in this document. Any other statements in this submittal are provided for information purposes and are not considered to be regulatory commitments. Please direct questions regarding these commitments to Mr. George Wrobel, License Renewal Project Manager at (585) 771-3535.

REGULATORY COMMITMENT		DUE DATE
C-RAI 4.2-1	Revise Surveillance Capsule Withdrawal Schedule and implement operating restrictions when last capsule is withdrawn.	RFO 2005 RFO 2009
C-RAI 3.6-1	Perform two SITs at design pressure during period of extended operation.	2009-2029
C-RAI B2.1.3-3(1)	Re-examine liner and restore thickness if below acceptance criteria.	2005
C-RAI 3.5-8	Include measurement of voltage between reference cells and rock anchors into PSPM program.	Prior to 2005

ATTACHMENT 1: CLARIFICATION RESPONSES

C-RAI 4.2.1 -1

The analysis for pressurized thermal shock in attachment 4 in the June 10, 2003, response for additional information letter is different than the evaluation in Section 4.2 of the LRA. Does this analysis supercede the evaluation documented in Section 4.2 of the LRA? If it does, UFSAR Section A3.1.2 needs to be revised. The applicant is requested to confirm the analysis documented in the June 10, 2002, letter supercedes the evaluation in the LRA and to provide a updated UFSAR Section A3.1.2.

Response

This analysis does supersede the evaluation documented in Section 4.2 of the LRA, in that a fluence of $5.01 \text{ E} + 19 \text{ n/cm}^2$ was used (equivalent to about 54 EFPY), rather than the $4.85 \text{ E} + 19 \text{ n/cm}^2$ used in the LRA.

We also noted a clerical error in the June 10 analysis - a margin term of 48.3°F was used in the Regulatory Guide 1.99, Rev. 2, Position 1.1 calculation, rather than 56°F . This results in an ART of 290.23°F , not 282.53°F .

The selected ART is still 270.6°F , based on Regulatory Position 2.1.

Section A3.1.2 of the application is being revised per this response. The second paragraph should read: "The results of the revised PTS analysis for the limiting material have been reviewed for compliance with 10CFR50.61. The methodology used in the PTS analysis for the beltline weld is based on the projected neutron fluence at the end of the period of extended operation, relying on plant-specific surveillance data to calculate ART, per position 2.1 of Regulatory Guide 1.99, Rev. 2. This analysis has been projected to the end of the period of extended operation, in accordance with 10CFR54.21(c)(1)(ii), and found to be acceptable.

Also, Section 4.2.1 of the LRA is being revised. The conclusion should state:

The results of the revised PTS analysis for the limiting material have been reviewed for compliance with 10CFR50.61. The methodology used in PTS analysis is based on the projected neutron fluence at the end of the period of extended operation and may rely on plant-specific surveillance data to calculate RT_{PTS} . Plant-specific surveillance data was used for the circumferential weld. Generic data calculated in accordance with Regulatory Guide 1.99, Rev. 2, Position 1.1 was used for the shells.

Table 4.2-1 Values of RT_{PTS} at EOL - Ginna RPV Beltline Materials

Material	Heat Number	Inner Surface Fluence E19 n/cm ²	Initial RT_{NDT} °F	Margin °F	Chemistry Factor °F	Inside Surface Fluence Factor	RT_{NDT} °F	RT_{PTS} °F
Intermediate Shell	125S255VA1	4.85	20	34 ¹	44 ¹	1.396	61.4	11.54
Lower Shell	125P666VA1	4.85	40	34 ¹	31 ¹	1.396	43.3	117.3
Circumferential Weld	61782/ SA-847	5.01	-4.8	48.3 ²	161.9 ²	1.403	227.1	270.6

¹ Regulatory Guide 1.99, Rev. 2, Position 1.1

² Regulatory Guide 1.99, Rev. 2, Position 2.1

The RT_{PTS} values for the intermediate and lower shell forgings remain below the NRC screening criterion of 270°F and the RT_{PTS} value for the beltline circumferential weld (SA-847) remains below the NRC screening criterion of 300°F at EOL. The analysis associated with PTS has been projected to the end of the period of extended operation and is consistent with 10CFR54.21(c)(1)(ii).

C-RAI 4.7.6 -1

In response to F-RAI 4.7.6-1 in the May 23, 2003, letter the applicant described the UFSAR Supplement to for the RCP flywheel TLAA and has semi-committed to update the UFSAR. Since the applicant has not updated the UFSAR, provide a updated UFSAR Section for review.

Response

The information provided in the May 23, 2003 response was intended to be the updated UFSAR section.

C-RAI 4.2.2 -1

Since the studs are fabricated with a specified minimum yield strength of 105 Ksi, it is possible that they could be heat treated to a maximum tensile strength limited greater than 1,172 MPa (170 ksi) and could be susceptible to SCC. This aging effect is identified in GALL item A2.1-c in NUREG-1801. This GALL item identifies Chapter XI.M3, "Reactor Head closure Studs" program as the GALL program acceptable for mitigating this aging effect. This program relies on ASME Code Section XI, Subsection IWB to monitor and detect this aging effect. Preventive measures identified in the GALL program include avoiding the use of metal-plated stud bolting to prevent degradation due to corrosion or hydrogen embrittlement and using manganese phosphate or other acceptable surface treatments and stable lubricants (RG 1.65).

Verify whether metal-plated stud bolting, manganese phosphate or other acceptable surface treatments, and stable lubricant was used/or the applicant is to provide the information.

Response

The reactor vessel closure studs are not plated with a metal coating. The studs were "Parkerized", which is a process for producing a manganese phosphate surface coating on steels. The lubricant used on the studs is N-7000, which is a stable, high-purity metal-free anti-seize lubricant suitable for use up to 2400°F.

C-RAI 4.2 -1

The current capsule withdrawal schedule is to withdraw one of the capsules during the 2003 refueling outage. At that time, the capsule will have received a fast neutron fluence of $5.05E19$, more than the projected dose at 60 years of $4.85E19$. Since Ginna has performed, and submitted to the NRC, a reactor vessel equivalent margins analysis, they indicated that they do not plan on testing that capsule. In addition, the current plan is to leave one capsule in the reactor vessel until about 2009, at which point it will have received a fast neutron fluence equivalent to 80 years of operation.

a) Since item 6 in GALL XI.M31 indicates the applicant is to withdraw one capsule at an outage in which the capsule receives a neutron fluence equivalent to the 60-year fluence and recommends that the applicant test the capsule in accordance with the requirements of ASTM E 185, the staff believes the capsule withdrawn during the 2003 refueling outage should be tested. Confirm whether the capsule will be tested during the current outage; if not, justify this deviation from GALL.

b) Item 7 in GALL XI.M31 indicates applicants without in-vessel capsules during the period of extended operation should use alternative dosimetry to monitor neutron fluence during the period of extended operation. Since the last capsule is to be removed in 2009, will Ginna have capsules within the vessel that could be removed and tested during the license renewal period? If they will not have capsules in the RPV during the license renewal period, what alternative dosimetry will be utilized during the period of extended operation to monitor neutron fluence?

Response

a) Our current (revised) schedule is to withdraw the next surveillance capsule during the 2005 refueling outage. It will have received a fluence of approximately $5.25 E + 19$ n/cm², which is greater than the projected EOL fluence of $4.85 E + 19$. ASTM E-185, Table 10, footnote E suggests that a fifth capsule be withdrawn when the fluence is not less than once or greater than twice the peak EOL vessel fluence. Capsule N will meet that criterion. Footnote E further states that this capsule may be held without testing following withdrawal. The Ginna Reactor Vessel Surveillance Program, though taking exception to NUREG-1801, is consistent with ASTM E-185.

As noted in our May 13, 2003 response, RG&E has requested Framatome to perform an Equivalent Margins Analysis for the Ginna reactor vessel, out through the period of extended operation. This fracture mechanics analysis confirmed that even though the USE might fall below 50 ft-1b, significant margin exists for the Ginna reactor vessel. Testing of the surveillance capsule coupons would not provide significant additional information.

b) Item 7 in GALL X1.M31 is not applicable to Ginna - we will be using the guidance provided in item 6 of that document.

When all surveillance capsules have been removed (~2009), operating restrictions will be established to ensure that the plant is operated under conditions to which the surveillance capsules were exposed and the exposure conditions of the reactor vessel will be evaluated to ensure that they continue to be consistent with those used to project the effects of embrittlement to the end of license. If the reactor vessel exposure conditions (neutron flux, spectrum, irradiation temperature, etc.) are altered, then the basis for the projection to 60 years is reviewed; and, if deemed appropriate, an active surveillance program will be re-instituted. Any changes to the reactor vessel exposure conditions and the potential need to re-institute a vessel surveillance program will be discussed with the NRC staff prior to changing the plant's licensing basis.

C-RAI 3.2.2 -1

The applicant indicates that credit is taken for the thimble tube inspections performed under the Thimble Tube Inspection Program as managing cracking due to SCC of the guide tubes. Details of these inspections including scope, examination method, acceptance criteria, and examination frequencies are included in the Thimble Tube Inspection Program description in Section B2.1.36 of the LRA. All thimble tube inspections are performed by personnel qualified in accordance with the requirements of ASME Section XI, Article IWA-2300, SNT-TC-1A, and ANSI/ASNT CP-189. Since the OD surface of the thimble tubes is exposed to the same environment as the ID surface of the guide tube and both components are fabricated from stainless steel they would both be susceptible to SCC. The Thimble Tube Inspection Program, as described in Section B2.1.36 of the LRA, is for detection of wear, not SCC. In order for the Thimble Tube Inspection to be utilized for detection of SCC in the guide tube, the Thimble Tube Inspection Program must be modified to include inspection for SCC. The staff requests that the applicant revise the Thimble Tube Inspection Program and the associated Ginna inspection procedures, as discussed in the response to F-RAI 3.2.2-1.

Response

The Thimble Tube Inspection Program has been revised to include cracking due to SCC as an aging effect requiring management for the thimble tubes and guide tubes.

Section A2.1.25 is being revised by adding the following sentence:

"Since ID surfaces of the BMI tubes are exposed to the same environment as the OD surfaces of the thimble tubes, this program is also credited for detecting and managing cracking due to SCC of the thimble tubes and guide tubes."

Section B2.1.36.4 is being revised by adding the following sentence:

"The aging effects which are detected by periodic eddy current testing of the thimble tubes are loss of material due to fretting wear and cracking due to SCC."

C-RAI 3.6-1

The 1993-94 correspondence, cited by the applicant, is related to the water damage pointed out by the staff. The applicant had performed structural analysis, and performed a Structural Integrity Testing (SIT) to confirm the continuing behavior or the containment. The staff agrees with the applicant that only indirect aging management could be performed for the three items in the lower part of the containment. The staff's expectation is that the applicant commit to perform two or three SITs during the extended period of operation. SIT could be performed at the peak calculated pressure that would demonstrate conformation with the expected behavior of the lower part of the containment. SIT measurements would consist of radial and vertical deformations similar to the measurements taken during initial and subsequent SITs, and visual observations during and after the test. The comparison will allow the applicant to detect significant deviation from the containment expected behavior.

Response

RG&E agrees to perform two structural integrity tests during the period of extended operation at the containment design pressure. We will perform these SITs in conjunction with the ILRTs, during which containment is also pressurized to design pressure. Based on our current schedule (10-year interval between ILRTs), these SITs would be scheduled to be conducted in 2015 and 2026. If Ginna Station were to be granted a 15-year duration interval between ILRTs, the SITs would be scheduled to be conducted in 2011 and 2026.

C-RAI B2.1.3-3(1)

Provide the acceptance criteria for liner degradation, when you (the applicant) will repair and restore the degraded liner before coating.

Response

Examinations of the Containment liner will be performed at Ginna Station in accordance with the requirements of ASME Section XI, Subsection IWE, Paragraph IWE-3512. Ultrasonic (UT) thickness measurements that reveal material losses exceeding 10% of the nominal Containment liner wall thickness, or material loss that is projected to exceed 10% of the nominal Containment liner wall thickness prior to the next examination, will be documented. Such areas may be accepted by engineering evaluation or corrected by repair or replacement in accordance with Paragraph IWE-3122. If either the thickness of the liner is reduced by no more than 10% of the nominal plate thickness or the reduced thickness can be shown by analysis to satisfy the

minimum design requirements, then such areas are acceptable by engineering evaluation.

The area of the liner exhibiting degradation that was discovered during previous inspections will be re-examined in 2005 and thereafter on a three-year frequency. The minimum required thickness of the Containment liner at Ginna Station has been determined by engineering analysis (and documented in EWR 5190) to be 0.281". Repair activities to restore the liner to its nominal thickness will be taken when the liner thickness reaches .300", or is expected to reach .300" before the next scheduled examination.

C-RAI 3.5-8

Please develop an aging management program for the periodic tests performed to measure the voltage between reference cells and the containment liner, tendons and rock anchors.

Response

This activity will be included in the Periodic Surveillance and Preventive Maintenance (PSPM) Program.

Section A2.1.17 will be modified by adding the words "periodic testing, as well as" between "provides for" and "evaluation of" in the third sentence.

Section B2.1.23. will be modified by adding the words "or by review of test results" at the end of the first sentence.

C-RAI B2.1.21-1

Please provide additional detail regarding response (5) to the One Time Inspection Program

Response

Table 3.4-2 of the LRA, line numbers (336) and (338) refer to the Reactor Make-up Water Storage Tank, which is a cylindrical carbon steel tank mounted vertically with a flat bottom that rests directly on the concrete floor. The interior surfaces of the tank are coated with an epoxy paint. The water stored in the tank is demineralized, oxygenated water. This tank was drained in 2000 for removal of the flexible rubber bladder. A thorough inspection of the interior of the tank was performed at that time. The inspection scope included visual examination of the interior surfaces of the tank, and ultrasonic thickness measurements of the tank bottom. The coating on the interior surfaces of the tank was in excellent condition, with no evidence of blistering, peeling, flaking, or substrate corrosion on the walls or bottom. The thickness measurements indicated no evidence of wall loss due to corrosion of the tank bottom. Based on the results of this inspection, at which time the tank had been in service for 30 years, there is reasonable assurance that the effects of aging will be managed by continued implementation of the Water Chemistry Control program at Ginna Station such that the intended function of the reactor make-up water storage tank will be maintained during the period of extended operation.

Table 3.4-2 of the LRA, line numbers (204, (205), (388) and (390) refer to carbon steel piping and valves exposed to demineralized water in the Treated Water System. The piping consists of 3/4" and 3" nominal diameter piping in Containment and the Auxiliary Building. There are three valves installed in the system, two manual valves and one air-operated valve. A review of plant-specific operating experience revealed no incidents or events related to age-related degradation of these components. Consequently, continued implementation of the Water Chemistry Control program, as well as a one-time inspection for verification of the effectiveness of water chemistry controls, provides reasonable assurance that the effects of aging will be managed such that the intended function of these components will be maintained during the period of extended operation. If the one-time inspection reveals evidence of age-related degradation, then appropriate corrective action will be taken and the components included in the scope of the periodic Surveillance and Preventive Maintenance program.

C-RAI B2.1.21-2

Please augment the UFSAR description of the One-Time Inspection Program, consistent with NUREG-1800.

Response

Section A2.1.15 should be modified to incorporate information provided in Tables 3.1-2, 3.2-2, 3.3-2, and 3.4-2 of NUREG-1800, as follows:

"For example, to verify the effectiveness of the water chemistry control program, a one-time inspection of small bore Class 1 piping and welds, using suitable techniques at the most susceptible locations, will be performed. Also, a one-time inspection of internal surfaces of carbon steel components using suitable techniques at the most susceptible locations will be performed to ensure unacceptable corrosion is not occurring.

To verify the effectiveness of the fuel oil program, a one-time inspection and thickness measurement of the diesel generator fuel oil tank will be performed".

This response supersedes our previous response to this RAI.

ATTACHMENT 2

Completion dates are being provided for commitments made in Enclosure 4 of our License Renewal Application submittal letter:

1. Submit a new pressure-temperature limit curves. **Due December 2004.**
2. Implement a Fatigue Monitoring Program confirm that the number of operating cycles (causing fatigue) are fewer than the plant design cycles. **Due June 2004.**
3. Provide an assessment of the fatigue usage for the nuclear sampling system B.31.1 piping, for the period of extended operation. **Completed.**
4. Provide a baseline NDE for the pressurizer surge line by inspecting all circumferential welds, and develop methodology to employ NRC-approved augmented ISI for pressurizer surge line or recalculate to determine acceptable CUF, or repair/replace surge line or subcomponents as necessary. **Completed reanalysis.**
5. Complete environmental qualification calculations to extend the qualified life of EQ components from 40 to 60 years, for those components using the TLAA criteria of 10 CFR 54.21 (c) (ii). **Due December 2004.**
6. Retension 23 containment tendons as part of the 2005 tendon testing program. **Due May 2005.**
7. Perform one-time inspections of selected plant equipment to verify that current plant aging management programs are effective in managing the effects of aging. **Due prior to September 2009.**
8. Enhance boric acid corrosion surveillance program to include all susceptible components (e.g., carbon/low alloy steel, copper) potentially exposed to boric acid leaks. **Completed.**
9. Develop a program to periodically assess the condition of non-EQ cables, in adverse localized environments. **Completed.**
10. Replace or test a representative sample of fire water system sprinklers that have been in service for 50 years. **Due prior to 2016.**
11. Develop a reactor vessel head penetration inspection program, in concert with industry initiatives. **Ongoing initiative with NEI and MRP.**
12. Participate with industry in helping to develop augmented inspection techniques to detect fine cracks and other changes in dimension in non-bolted components of the reactor vessel internals. **Ongoing initiative with NEI and MRP.**

13. Enhance structural monitoring program to include all structures within the scope of license renewal, and provide additional guidance for detecting aging effects. **Completed.**
14. Enhance systems monitoring program to include all systems within the scope of license renewal, and provide additional guidance for detecting aging effects. **Due June 2004.**