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SUBJECT: Forwards addl info in response to AR Johnson 930924 ltr re Generic Ltr 92-01, "Reactor Vessel Structural Integrity, 10CFR50.54(f)."

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November 29, 1993

U.S. Nuclear Regulatory Commission
Document Control Desk
Attn: Allen R. Johnson
Project Directorate I-3
Washington, D.C. 20555

Subject: Reactor Vessel Integrity
Response to Request for Additional Information
R.E. Ginna Nuclear Power Plant
Docket No. 50-244

Ref.(a): Letter from A. R. Johnson (NRC) to R. C. Mecredy (RG&E), "Reactor Vessel Structural Integrity - Request for Additional Information," dated September 24, 1993

(b): Letter from R. C. Mecredy (RG&E) to A. R. Johnson (NRC), "Reactor Vessel Structural Integrity, 10CFR50.54(f) Response to Generic Letter 92-01, Revision 1," dated July 2, 1992

Dear Mr. Johnson:

The purpose of this letter is to provide additional information, as requested by reference (a), regarding the Rochester Gas and Electric response to Generic Letter 92-01, reference (b). The requested information is provided in Table A to this letter. Table B provides preliminary Charpy upper-shelf energy (USE) data obtained from the latest (1993) surveillance capsule tests which demonstrate that the Ginna Station reactor vessel continues to exceed the 50 ft. lb. requirement of 10CFR50, Appendix G. The final results for capsule "S" will be provided when available.

Very truly yours,

Robert C. Mecredy

Robert C. Mecredy

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Attachments

xc: Mr. Allen R. Johnson (Mail Stop 14D1)
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Ginna Senior Resident Inspector

TABLE A

Additional information in response to NRC-RAI dated September 24, 1993:

A. Unirradiated C_vUSE Values

1. Forging / C_vUSE (ft. lb)

Forging	C_vUSE (ft. lb)
125P666VA1	114 (NOTE 1)
123P118VA1	117 (NOTE 2)

2. Weld / C_vUSE (ft-lb)

Weld	C_vUSE (ft-lb)
SA-1101	70 (NOTE 3)
SA-848	70 (NOTE 3)

NOTE 1: The Charpy impact data reported in the Materials Test Report (Bethlehem Steel Corp., Report of Tests, Report No. 2203, October 24, 1966) for forging 125P666VA1 indicates an initial C_vUSE values of 176 ft-lb. The specimens were oriented such that the break is in the strong direction. This value is below the strong direction C_vUSE value (183 ft-lb) reported in WCAP-7254 and WCAP-10086 for the forging 125P666VA1 surveillance material. Therefore, for conservatism 176 ft-lb will be used to represent the C_vUSE value in the strong direction for the beltline forging 125P666VA1. In accordance with the Standard Review Plan, Section 5.3.2, 65% of the strong direction C_vUSE value may be taken as the weak direction C_vUSE value. Therefore, the weak direction C_vUSE value is 114 ft-lb (65% of 176 ft-lb). This value is well in excess of the minimum 75 ft-lb that is initially required for reactor vessel beltline materials in accordance with 10CFR50, Appendix G.

NOTE 2: The Charpy impact data reported in the Materials Test Report (Bethlehem Steel Corp., Report of Tests, Report No. 679, May 24, 1966) for forging 123P118VA1 indicates an initial C_vUSE value of 180.7 ft-lb. The specimens were oriented such that the break is in the strong direction. In accordance with the Standard Review Plan, Section 5.3.2, 65% of the strong direction C_vUSE value may be taken as the weak direction C_vUSE value. Therefore, the weak direction C_vUSE value is 117 ft-lb (65% of 180.7 ft-lb). This value is well in excess of the minimum of 75 ft-lb that is initially required for reactor vessel beltline materials in accordance with 10CFR50, Appendix G.

NOTE 3: The unirradiated C_vUSE for SA-1101 and SA-848 is 70 ft-lb. This value is taken from Table 3-5 of BAW-1803; it is rounded from 69.7 ft-lb. This value was statistically derived from the entire population of Linde 80 welds and

is considered to be the most representative value obtainable, since it was obtained using valid and applicable information for Linde 80 weld material. Analyses have demonstrated that material with this low C_vUSE value will provide margins of safety against fracture toughness. These analyses are documented in reports BAW-2192P and BAW-2178P which have been submitted to the NRC for acceptance.

B. Copper Content for Forging 123P118VA1

The copper content for forging 123P118VA1 is not available. This forging was fabricated to the ASME SA-336 specification which does not require the reporting of a copper composition. Therefore, as specified in Regulatory Guide 1.99, Revision 2, a copper content of 0.35% is assumed. This of course is greatly conservative since typical copper content for reactor vessel forging materials is in the range of 0.01 to 0.10%.

C. Fluence Values

Capsule	Fluence (n/cm^2) (reported in response to GL 92-01)
V	6.53E18
R	1.02E19
T	1.78E19

The fluence values specified in the response to GL 92-01, Revision 1, were obtained from BAW-1803, Revision 1, which restate the revised fluence values reported in WCAP-11026. To date, four surveillance capsules have been withdrawn from the R. E. Ginna reactor and the dosimetry from these irradiations has been evaluated by Westinghouse using current evaluation techniques. Two of these capsules were withdrawn at the conclusion of Fuel Cycles 2 and 3 in Fall 1972 and Spring 1974, respectively; while the third capsule was withdrawn following the completion of Fuel Cycle 9 in Spring 1980. These revised fluence values, being larger than those documented in the earlier surveillance reports, are considered more conservative. The fourth capsule, Capsule S, was removed in Spring 1993 and is currently being analyzed. Preliminary results are provided in Table B.

TABLE B

Latest weld capsule surveillance test results:

Capsule	C_vUSE(ft-lb)	Fluence (n/cm^2)
T	55.4	1.78E19
S (preliminary)	54.5	3.87E19