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SUBJECT: Provides pressurized thermal shock assessment for reactor pressure vessel limiting circumferential weld for review & approval.

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ROBERT C. MECREDY
Vice President
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October 11, 1995

U.S. Nuclear Regulatory Commission
Document Control Desk
Attn: Allen R. Johnson
Project Directorate I-1
Washington, D.C. 20555

Subject: Pressurized Thermal Shock
Assessment for Ginna Reactor Vessel
R.E. Ginna Nuclear Power Plant
Docket No. 50-244

Ref.(a): Letter from Allen R. Johnson (NRC), to Robert C. Mecredy (RG&E), "Summary of Meeting with Rochester Gas and Electric Corporation on May 16, 1995 - Pressurizer Thermal Shock Assessment of Reactor Vessels," dated June 5, 1995.

Dear Mr. Johnson:

The purpose of this correspondence is to provide Rochester Gas and Electric Corporation's (RG&E) pressurized thermal shock (PTS) assessment for the R.E. Ginna reactor pressure vessel (RPV) limiting circumferential weld for your review and approval. The assessment, as presented in the attached, is provided in response to the referenced letter and is based on the guidance provided in 10CFR50.61 and Regulatory Guide 1.99 revision 2. As provided for by this guidance, the RG&E assessment includes the use of the results of the RPV surveillance capsule program. As the attached PTS determination shows, the Ginna RPV can be expected to operate past the end of license (EOL) of 2009 without exceeding the PTS screening criterion of 10CFR50.61 for the limiting case weld.

Very truly yours,

Robert C. Mecredy

Robert C. Mecredy

REJ\393

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Ginna Senior Resident Inspector

ATTACHMENT

R.E. Ginna Reactor Vessel EOL PTS Assessment

I. Discussion

The purpose of this assessment is to show that the expected reference temperature PTS (RT_{PTS}) at EOL for the Ginna RPV limiting weld remains below the screening criteria when using surveillance program data. The limiting material for the Ginna RPV is the intermediate to lower shell weld having weld wire heat number 61782. The material used for the surveillance capsule program is weld wire SA-1036 having material heat number 61782 which has been determined to be acceptable for use as a surveillance material[6]. The determination of the reference temperature for RT_{PTS} for the RPV-PTS assessment uses the 10CFR50.61(b)(3) plant specific surveillance program method which includes the surveillance data adjustments per Regulatory Guide 1.99 revision 2 position 2, where:

(1) $RT_{PTS} = I + M + \Delta RT_{NDT}$[8]

(2) $I =$ initial RT_{NDT}[8]

(3) $M =$ margin $= 2\sqrt{\sigma_I^2 + \sigma_\Delta^2}$ when using surveillance data[4]

(4) $\Delta RT_{NDT} =$ reference temperature shift[8]

II. Material Values Used

A. RPV material heat 61782

1. I = the mean of the test values =

$$\frac{\sum_{i=1}^{n} \chi i}{n} = \quad \text{where: } i_1 = -1°F^{(1)},\ i_n = -38°F^{(5)},\ n = 2$$

$\Rightarrow I = -19.5°F$ and $\sigma_I = 18.5°F$

2. fluence @ EOL = 3.68E19[2]

3. Chemistry[1]
   Cu = .25
   Ni = .54

4. CF[3]:
   167.6°F

5. $M = 2\sqrt{\sigma_I^2 + \sigma_\Delta^2}$ where $\sigma_I = 18.5°F$ (II.A.1) and $\sigma_\Delta = 14°F$[4]
   $\Rightarrow M = 46.4°F$

B. Surveillance Capsules Weld SA 1036

1.

Capsule No.[2]	Fluence(B)[2]	ΔRT_{NDT}[2]
V	5.56E18	140°F
R	1.15E19	165°F
T	1.97E19	150°F
S	3.87E19	205°F

2. Mean Value Chemistry[7][2][9]:

   Cu = .214
   Ni = .505

3. CF[3]:

   150.9°F

III. Calculations

A. Chemistry factor ratio[4]:

CF of RPV (II.A.4 above) ÷ CF of capsule (II.B.3 above)

$\Rightarrow$ Ratio = $\frac{167.6}{150.9}$ = 1.11

B. Best fit CF[4]

1. Summing adjusted capsule ΔRT_{NDT}

Capsule	ΔRT_{NDT} (II.B.1)	X	Ratio (III.A)	Adjusted ΔRT_{NDT}
V	140	X	1.11	155.4
R	165	X	1.11	183.2
T	150	X	1.11	166.5
S	205	X	1.11	227.6

2. Determine sum of fluence factor x adjusted ΔRT_{NDT}

where fluence factor (ff) = $f^{(0.28 - 0.10 \log f)}$

Capsule	FF	Adjusted ΔRT_{NDT}	Product
V	0.836	155.4	129.9.
R	1.039	183.2	190.3
T	1.185	166.5	197.3
S	1.349	227.6	307.
		sum =	824.5

3. Summing squares of fluence factors (FF) for capsules

using fluence factor values from II.B.1:

capsule	ff	$(ff)^2$
V	0.836	0.699
R	1.039	1.080
T	1.185	1.405
S	1.349	1.820
	sum =	5.004

4. Sum of adjusted ΔRT_{NDT} ÷ sum of squares of fluence

824.5 (III.B.2) ÷ 5.004 (III.B.3) = 164.8°F

C. ΔRT_{NDT} for surveillance fit @ EOL

EOL ΔRT_{NDT} = (CF) $f^{(0.28 - 0.10 \log f)}$ where f = 3.68

$\Rightarrow$ EOL ΔRT_{NDT} = 164.8°F (III.B.4) x $3.68^{(0.28 - 0.10 \log 3.68)}$

= (164.8°F)(1.338) = 220.5

D. RT_{PTS} @ EOL

RT_{PTS} = I + M + ΔRT_{NDT}

= −19.5°F (II.A.1) + 46.4°F (II.A.5) + 220.5 (III.C)

= 247.4°F

IV. Results

The RT_{PTS} value at EOL for the R.E. Ginna vessel is 247.4°F. This is 52.6°F below the PTS screening criterion described in 10CFR50.61.

REFERENCES:

(1) BAW 1803 Revision 1, "Correlations for Predicting the Effects of Neutron Radiation on Linde 80 Submerged - Arc Welds."

(2) WCAP-13902, "Analysis of Capsule S from the Rochester Gas and Electric Corporation R.e. Ginna Reactor Vessel Radiation Surveillance Program," dated December 1993.

(3) Code of Federal Regulations, 10CFR50.61, "Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events."

(4) Regulatory Guide 1.99, revision 2, "Radiation Embrittlement of Reactor Vessel Materials."

(5) BAW 1920P, "Analysis of Capsule DB1-LG1," October 1986.

(6) Letter to Roger W. Kober from Morton B. Fairtile, "Safety Evaluation by the Office of Nuclear Reactor Regulation Related to Amendment No. 15 to Facility Operating License No. DPR-18 Rochester Gas and Electric Corporation R.E. Ginna Nuclear Power Plant Docket No. 50-244," dated June 12, 1986.

(7) BAW-1500, "Chemistry of 177-FA B& W Owner's Group Reactor Vessel Beltline Welds," September 1978.

(8) 10CFR50.61, "Fracture toughness requirements for protection against pressurized thermal shock events."

(9) BAW-2121P, "Chemical Composition of B&W Fabricated Reactor Vessel Beltline Welds," April 1991.