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Project Directorate I-3

SUBJECT: Forwards info re reactor vessel issues,per 900305 telcon concerning change of license expiration date.

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ROCHESTER GAS AND ELECTRIC CORPORATION • 89 EAST AVENUE, ROCHESTER, N.Y. 14649-0001

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March 28, 1990

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

Subject: Change of License Expiration Date
Additional Information
R.E. Ginna Nuclear Power Plant
Docket No. 50-244

Reference: (1) Letter from Bruce Snow (RG&E) to Carl Stahle (NRC), dated January 19, 1988 of same subject
(2) Letter from Robert Smith (RG&E) to Allen Johnson (NRC), dated October 5, 1989 of same subject
(3) Telephone Conference of March 5, 1990 concerning issues related to change in license expiration

Dear Mr. Johnson:

On March 5, 1990, a telephone conference (Reference 3) was held with Allen Johnson (NRC), John Tsao (NRC), Peter Nagata (EG&G), John Smith (RG&E), George Wrobel (RG&E), Mike Saporito (RG&E), William Galloway (RG&E), Robert Eliasz (RG&E) and John Jorgensen (RG&E). During the course of the conversation, the NRC and your contractor asked a number of questions which centered around reactor vessel issues. The attached data is being supplied to you and directly to your EG&G contractor (per your instructions) in an attempt to address your concerns.

Very truly yours,

Robert C. Mecredy

Robert C. Mecredy
Division Manager
Nuclear Production

JRJ\096
Attachments

A001 1/1 Cut No .P340957207

xc: Mr. Allen R. Johnson (Mail Stop 14D1)
Project Directorate I-3
Washington, D.C. 20555

U.S. Nuclear Regulatory Commission
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Idaho Falls, ID 83415

Ginna Senior Resident Inspector

LICENSE AMENDMENT - CHANGE OF EXPIRATION DATE

Questions from March 5, 1990 NRC/RG&E Telecon

1. Question

RG&E's October 5, 1989 submittal has only one sentence in Attachment B regarding the issue of Reactor Vessel Surveillance. In particular, the NRC is interested in how the requirements of 10CFR50, Appendix H are being addressed.

Response

The purpose of RG&E's Material Surveillance Program is to monitor changes in the fracture toughness properties of ferritic materials in the reactor vessel beltline region resulting from exposure of these materials to neutron irradiation and the thermal environment. The information from the Material Surveillance Program is used as input to the 10CFR50, Appendix G Fracture Toughness Requirements. As stated in Section 5.3.3.2 of the Ginna UFSAR and WCAP-7254, the Material Surveillance Program is designed to meet the requirements of Appendix H and ASTM E-185-73. It consists of six surveillance capsules positioned in the reactor vessel between the thermal shield and the vessel wall. The vertical center of each capsule is opposite the vertical center of the core. Each capsule contains tensile, Charpy V-notch and wedge opening loading specimens from the forgings (heats 125P666 and 125S255), and weld metal and Charpy V-notch specimens from the heat affected zone material. The surveillance capsules contain dosimeter wires of copper, nickel and aluminum-cobalt. They also contain cadmium-shielded dosimeters of Neptunium-237 and Uranium-238. The dosimeters permit evaluation of the neutron flux seen by the various specimens.

The surveillance capsules can be removed when the vessel head is removed. The capsules can be replaced when the vessel internals are removed. Surveillance capsules are periodically removed and tested in accordance with Technical Specifications and test results are documented. Test results are analyzed, the shift in transition temperature is compared to the predicted shift and pressure-temperature limit curves are revised accordingly.

Capsules withdrawn after July 26, 1983 will be tested and results reported in accordance with the 1982 revision of ASTM E-185 as required by Appendix H requirements.

The Ginna capsule withdrawal schedule is in accordance with E-185 and Technical Specification Section 4.3.1.1 and is as follows:

Capsule	Time Removed for Testing
V	Removed in 1971
R	Removed in 1974
T	Removed in 1980
P	17EFPY at nearest refueling
S	Standby
N	Standby

The above capsule withdrawal schedule may change based on the Integrated Reactor Vessel Surveillance Program of Babcock and Wilcox in which RG&E participates. If such a change is requested, it will be covered by a separate submittal.

The results of reactor vessel material surveillance capsules V, R and T have been submitted to the NRC in accordance with ASTM E-185 requirements. The report numbers for these capsules are listed below:

Capsule	Report No.
V	FP-RA-1
R	WCAP 8421
T	WCAP 10086

2. Question

The NRC wishes to have a commitment that the Charpy upper shelf energy of the Ginna vessel is above 50 ft-lb. as required by 10CFR50, Appendix G.

Response

The beltline weld is the limiting material in the Ginna reactor vessel. Three capsules have been removed and tested and the Charpy upper shelf energy results for the weld material is summarized as follows:

Capsule V: Upper shelf energy decreased from 74.0 ft-lbs. to 50.8 ft-lbs. at a fluence of 4.9 x 10^{18} neutrons/cm^2 (E > 1 Mev).

Capsule R: Upper shelf energy decreased from 80.0 ft-lbs. to 50.0 ft-lbs. at a fluence of 7.6 x 10^{18} neutrons/cm^2 (E > 1 Mev).

Capsule T: Upper shelf energy decreased from 80.0 ft-lbs. to 55.0 ft-lbs. at a fluence of 1.75 x 10^{19} neutrons/cm^2 (E > 1 Mev).

In addition, reconstituted Charpy tests were performed on weld metal from Capsule T. The upper shelf energy decreased from 80.0 ft-lbs. to 55.0 ft-lbs. at a fluence of 1.75 x 10^{19} neutrons/cm^2 (E > 1 Mev).

Based on the surveillance capsule results to date, it can be concluded that the reactor beltline region material is not as sensitive to radiation as predicted and a saturation of radiation damage may be occurring.

In addition to the above, a 100% examination of the reactor vessel beltline weld was performed during the 1989 refueling outage with no recordable indications found. (See letter of May 4, 1989 from R.E. Smith (RG&E) to NRC concerning Reactor Vessel Examination of 1989.)

RG&E has joined the Babcock and Wilcox Owners Group (BWOG) which was formed to address the upper shelf toughness issue for all B&W fabricated reactor vessels with Linde 80 welds. The principal objective of this group is to assure the continued licensability of all eleven participants and their seventeen reactor vessels. The group will provide the data required of Appendix G, 10CFR50 to accomplish this objective.

In the event that the Charpy upper shelf energy falls below 50 ft-lbs. in the future, an analysis would be performed that would conservatively demonstrate, making appropriate allowances for all uncertainties, the existence of equivalent margins of safety for continued operation.

Finally, if there is no indication that an equivalent safety margin exists, then the reactor vessel may have to be thermally annealed to recover the fracture toughness of the material. Westinghouse and EPRI have been working on a program to demonstrate the feasibility of this type of procedure as documented in EPRI Report No. NP-6113-M, Thermal Annealing of an Embrittled Reactor Vessel.

3. <u>Question</u>

The NRC is interested in RG&E's response to Generic Letter 88-11, "NRC Position on Radiation Embrittlement of Reactor Vessel Materials and its Impact on Plant Operations".

<u>Response</u>

Our response was provided by letter dated January 23, 1989. In accordance with our commitments in that response, RG&E expects new heatup and cooldown curves to be implemented by May 1991.

4. <u>Question</u>

There is a discrepancy in WCAP-8421 concerning data for the Charpy upper shelf energy of forging 125666VA1 which is the lower shell forging. Table 5-6 on Page 5-13 lists an upper shelf energy of approximately 183 ft-lbs. while Table A-1 on Page A-4 lists an upper shelf energy of approximately 164 ft-lbs. Which is correct?

Response

The average value of approximately 183 ft-lbs. for forging 125P666 is derived from Figure 5-1 on Page 5-8 of the Capsule Report. Unirradiated values of 202 ft-lbs., 186 ft-lbs., and 162 ft-lbs. are averaged to give 183 ft-lbs. The average value of approximately 183 ft-lbs. for forging 125P666 is correct and is consistent with values reported in Capsules V and T, as well as Capsule R. Table A-1 on Page A-4 of Capsule R was apparently copied from the report of Capsule V which has the same incorrect value of approximately 164 on Page 45.

5. Question

Page 4-1 of Capsule R Report appears to have the lower and intermediate forging numbers reversed.

Response

The sentence from Page 4-1 reads in part ...intermediate and lower shell forgings (heats 125P666 and 125S255).... This sentence is confusing and does have the numbers in the wrong order. This should read ...intermediate forging (125S255) and lower forging (125P666VA1).... The correct forging numbers are shown on Ginna UFSAR Figure 5.3-2, which is attached.

6. Question

NRC requested a copy of Westinghouse Letter NSID WOG-37, dated November 16, 1982 to R. Mecredy regarding Ginna Reactor Vessel Material.

Response

Letter attached.

7. Question

What material is limiting?

Response

As stated in Section 5.3.1.2 of the Ginna UFSAR, the beltline weld, SA-847, is limiting. The location of this weld is shown on the attached Figure 5.3-2 of the Ginna UFSAR.

8. Question

Forging 125S255 has showed no shift in RT_{NDT} for Capsule R.

Response

There was a 0°F shift in RT_{NDT} for Capsule R, this was also true for Capsule V and Capsule T. RG&E can only speculate as to the reason for the 0°F shift, however, it appears that for some reason this material is insensitive to radiation.

Copy to A. Curtis
T. Meyer

Westinghouse Electric Corporation | **Water Reactor Divisions**

Nuclear Service Division

Box 2728
Pittsburgh Pennsylvania 15230

November 16, 1982

NSID/WOG-37

Dr. Robert Mecredy
Rochester Gas & Electric
89 East Avenue
Rochester, New York 14649

Dear Dr. Mecredy:

<u>Robert E. Ginna Reactor Vessel Material</u>

Attached is a complete specification of the base material and weld material for the Robert E. Ginna reactor pressure vessel. The source of this data is the original material certification supplied by Babcock and Wilcox with the vessel. This is the data forming the basis for the Westinghouse RT_{NDT} calculations.

Very truly yours,

Rick Muench

R. A. Muench, Manager
Westinghouse Owners Group

/elr

Attachment

cc: G. W. Dillon

ROBERT E. GINNA UNIT NO. 1

REACTOR VESSEL MATERIAL SURVEILLANCE PROGRAM

1.) The estimated maximum fluence (E > 1 Mev) at the inner surface of the reactor vessel wall as of March 31, 1977 is 5.26×10^{18} n/cm^2.

2.) The effective full power years (EFPY) of operation accumulated as of March 31, 1977 is 4.55 EFPY.

3.) Fabrication of the reactor vessel was performed by Babcock & Wilcox Co.

4.) a.) Sketch of the reactor vessel showing base material and welds in the beltline region is shown in Figure 1.

b.) Information on each of the welds in the beltline region is shown in Tables 1 through 4.

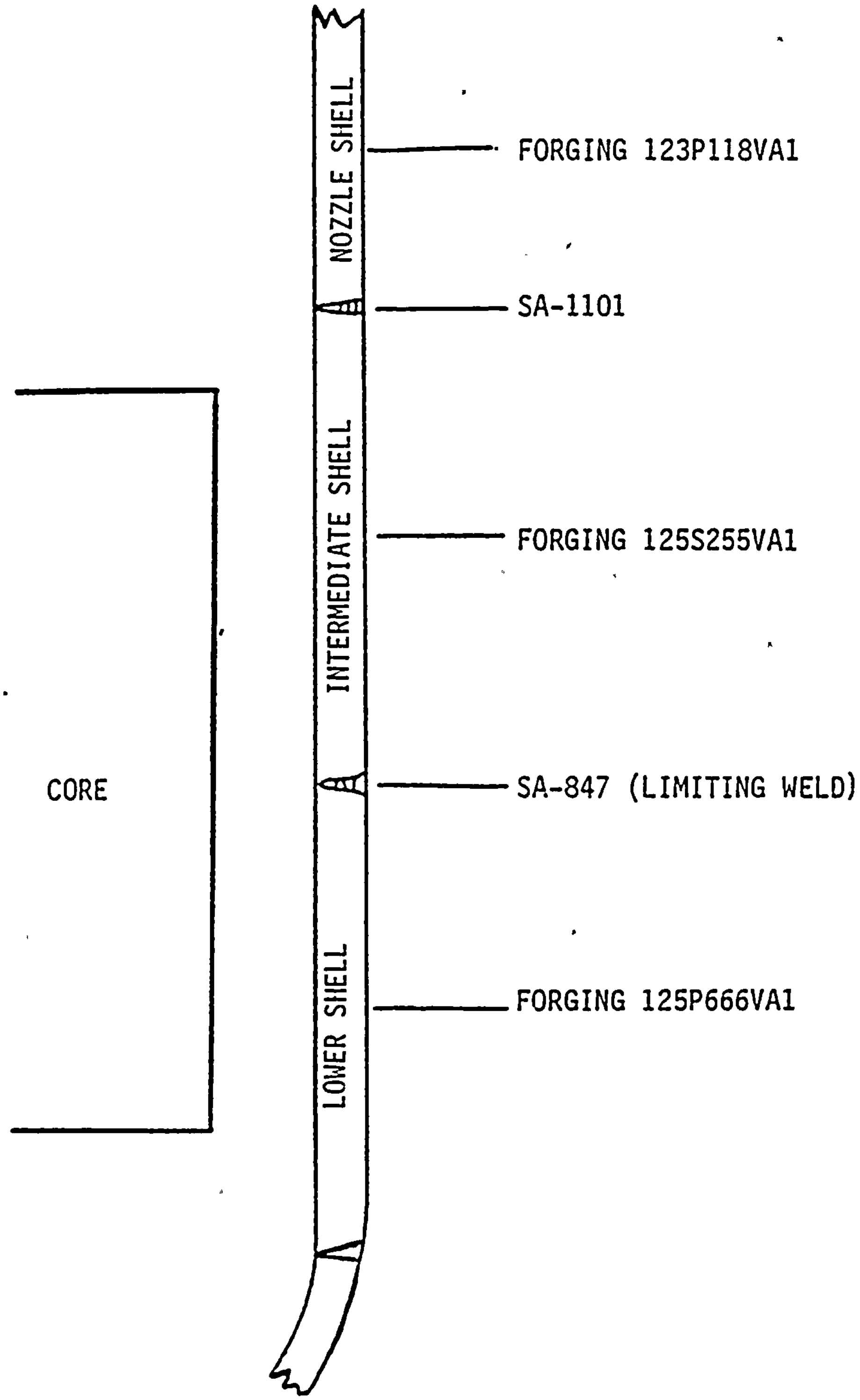
c.) Information on each of the shell forgings in the beltline region is shown in Tables 4 through 7.

5.) Information relative to weld and forging material included in the material surveillance program is shown in Tables 1 through 3 and 5 through 7.

(incomplete)

FIGURE 1

IDENTIFICATION AND LOCATION OF BELTLINE REGION MATERIAL
ROBERT E. GINNA UNIT NO. 1 REACTOR VESSEL



ROCHESTER GAS AND ELECTRIC CORPORATION
R.E. GINNA NUCLEAR POWER PLANT
UPDATED FINAL SAFETY ANALYSIS REPORT

Figure 5.3-2

Identification and Location of Beltline Region Material

TABLE 1

IDENTIFICATION OF REACTOR VESSEL BELTLINE REGION WELD MATERIAL

Weld Location	Weld Process	Weld Control No.	Weld Wire Type	Weld Wire Heat No.	Flux Type	Flux Lot No.	Post Weld Heat Treatment
Nozzle Shell to Inter. Shell	Submerged Arc	SA-1101	Mn-Mo-Ni	71249	Linde 80	8445	1100-1125°F-48 Hrs.-FC
Inter. Shell to Lower Shell	Submerged Arc	SA-847	Mn-Mo-Ni	61782	Linde 80	8350	1100-1125°F-48 Hrs.-FC
Surveillance Weld	Submerged Arc	SA-1036	Mn-Mo-Ni	61782	Linde 80	8436	1100°F-11-1/4 Hrs.-FC

TABLE 2

WELD MATERIAL CHEMICAL COMPOSITION

Weld Control No.	Weld Wire Type	Weld Wire Heat No.	Flux Type	Flux Lot No.	C	P	S	Mn	Si	Mo	Ni	Cr	Cu
SA-1101	Mn-Mo-Ni	71249	Linde 80	8445	.070	.021	.014	1.28	.52	.36	.57	.17	.21
SA-847	Mn-Mo-Ni	61782	Linde 80	8350	.082	.012	.012	1.34	.45	.39	.39	.06	.20
Surveillance Weld					.075	.012	.016	1.31	.59	.36	.56	.59	.23

(C through Cu: Weight Percent)

TABLE 3

MECHANICAL PROPERTIES OF WELD MATERIAL

Weld Control No.	Weld Wire Type	Weld Wire Heat No.	Flux Type	Flux Lot No.	T_{NDT} °F	Energy at 10°F ft-lb	RT_{NDT} °F	Shelf Energy ft-lb	YS ksi	UTS ksi	Elong. %	RA %
SA-1484	Mn-Mo-Ni	71249	Linde 80	8445	0*	45, 45, 46	0*	----	68.63	84.26	28.5	----
SA-1101	Mn-Mo-Ni	61782	Linde 80	8350	0*	58, 60, 36	0*	----	67.00	81.88	29.5	----
Surveillance Weld					0*	54, 66.5, 71**	0*	79.0	73.52	87.35	22.8	62.0

* Estimated based on NRC Standard Review Plan Section 5.3.2 and MTEB 5-2

** Energy at 60°F

TABLE 4

MAXIMUM END OF LIFE FLUENCE AT VESSEL WALL LOCATIONS

	Fluence (n/cm^2)
Nozzle Shell to Inter. Shell Weld	$\sim 2.0 \times 10^{18}$
Inter. Shell to Low Shell Weld	3.7×10^{19}
Nozzle Shell Forging 123P118VA1	$\sim 2.0 \times 10^{18}$
Inter Shell Forging 125S255VA1	3.7×10^{19}
Lower Shell Forging 125P666VA1	3.7×10^{19}

TABLE 5

IDENTIFICATION OF REACTOR VESSEL BELTLINE FORGING MATERIAL

Component	Forging No.	Heat No.	Material Spec.	Supplier	Heat Treatment: Austenitize	Heat Treatment: Temper	Heat Treatment: Stress Relief
Nozzle Shell	123P118VA1	123P118	A336	Bethlehem Steel	1550°F-11 Hrs-WQ	1220°F-22 Hrs-AC	1125°F-30 Hrs-FC
Inter. Shell	125S255VA1	125S255	A508 CL2	Bethlehem Steel	1550°F-15-1/2 Hrs-WQ	1210°F-18 Hrs-AC	1125°F-30 Hrs-FC
Lower Shell	125P666VA1	125P666	A508 CL2	Bethlehem Steel	1550°F-9 Hrs-WQ	1220°F-12 Hrs-AC	1125°F-30 Hrs-FC
Surveillance Forgings	125S255VA1	125S255	A508 CL2	Bethlehem Steel	1550°F-15-1/2 Hrs-WQ	1210°F-18 Hrs-AC	1100°F-11-1/4 Hrs-FC
	125P666VA1	125P666	A508 CL2	Bethlehem Steel	1550°F-9 Hrs-AC	1220°F-12 Hrs-AC	1100°F-11 Hrs-FC

TABLE 6

BELTLINE FORGING MATERIAL CHEMICAL COMPOSITION

Forging No.	Weight Percent: C	P	S	Mn	Si	Mo	Ni	Cr	Cu	V
123P118VA1	.19	.010	.009	.65	.23	.60	.69	.42	---	---
125S255VA1	.18	.010	.007	.66	.23	.58	.69	.33	.07	.02
125P666VA1	.19	.012	.011	.67	.20	.57	.69	.37	.05	.02

TABLE 7

MECHANICAL PROPERTIES OF BELTLINE FORGINGS

Forging No.	T_{NDT} °F	RT_{NDT} °F	Upper Shelf Energy ft-lb	YS ksi	UTS ksi	Elong. %	RA %	
123P118VA1	40	40*	117*	66.87	88.00	25.50	73.50	
125S255VA1	20	20*	106*	67.25	88.25	26.25	70.10	
125P666VA1	40	40*	114*	63.50	85.00	26.25	71.05	
125S255VA1	20	20*	91*	78.22	97.19	23.30	66.85	Surveillance Test Results
125P666VA1	40	40*	120*	62.72	83.65	26.35	70.75	

*Estimated Based on NRC Standard Review Plan Section 5.3.2 and MTEB 5.2