BAW-2260
OCTOBER 1995

RESPONSE TO GENERIC LETTER 92-01, REVISION 1, SUPPLEMENT 1, FOR VIRGINIA POWER'S NORTH ANNA UNITS 1 AND 2 BELTLINE MATERIALS AND SURRY UNITS 1 AND 2 ROTTERDAM BELTLINE WELD METALS

B&W NUCLEAR TECHNOLOGIES

BAW-2260
October 1995

Response to Generic Letter 92-01, Revision 1, Supplement 1 for Virginia Power's North Anna Units 1 and 2 Beltline Materials and Surry Units 1 and 2 Rotterdam Beltline Weld Metals

by

M. J. DeVan

BWNT Document No. 77-2260-00
(See Section 4 for document signatures.)

Prepared for

Virginia Power

Prepared by

B&W Nuclear Technologies, Inc.
Engineering and Project Services Division
3315 Old Forest Road
P. O. Box 10935
Lynchburg, Virginia 24506-0935

CONTENTS

List of Tables

Table

1.0 INTRODUCTION

This report provides a response to the Nuclear Regulatory Commission (NRC) Generic Letter 92-01, Revision 1, Supplement 1, for the Virginia Power's North Anna Units 1 and 2 beltline materials and Surry Units 1 and 2 beltline weld materials that were fabricated by Rotterdam Dockyard Company.

Generic Letter 92-01, Revision 1, Supplement 1, was issued by the NRC on May 19, 1995 and addressed to all holders of nuclear power plant operating licensees. The generic letter was issued to require the licensees to identify, collect, and report any new data pertinent to the analysis of structural integrity of their reactor vessels and assess the impact of that data on their reactor vessel integrity analyses relative to the requirements of current regulations.

2.0 STATEMENT OF RESPONSE

2.1 Part (1) to NRC Generic Letter 92-01, Revision 1, Supplement 1

Part (1) of the Generic Letter requires the Addressees to provide the following information:

> "a description of those actions taken or planned to locate all data relevant to the determination of RPV integrity, or an explanation of why the existing data base is considered complete as previously submitted;"

The North Anna Units 1 and 2 reactor vessels, and portions of the Surry Units 1 and 2 reactor vessels, were fabricated by the Rotterdam Dockyard Company. To date, the following sources of information have been identified and utilized in the evaluation of the North Anna Units 1 and 2 reactor vessel beltline materials and the Surry Units 1 and 2 Rotterdam weld materials:

- Plant-specific reactor vessel materials surveillance reports
- Virginia Power documentation on reactor vessel materials (i.e., vendor correspondence, reports, etc.)

Because several other U.S. nuclear reactor vessels were fabricated by the Rotterdam Dockyard Company, additional information on Rotterdam weld materials and weld material surrogates may exist for the weld wire heats used in the fabrication of the North Anna and Surry reactor vessels. To ensure that all relevant chemical and mechanical test data have been identified, confirmatory searches of all known data sources for these reactor vessel beltline materials will be reviewed. Potential sources of additional data include:

- Westinghouse Electric Company records
- Rotterdam Dockyard Company fabrication records
- Virginia Power internal documentation
- Industry data bases (i.e., PREP3, PR-EDB, RVID, etc.)

The portions of the Surry Units 1 and 2 reactor vessels that were not fabricated by Rotterdam Dockyard Company were fabricated by Babcock & Wilcox (B&W). The response to Part (1) of Generic Letter 92-01, Revision 1, Supplement 1, for the B&W-fabricated portions of the Surry Units 1 and 2 reactor vessels will be included in the forthcoming B&W Owners Group (B&WOG) Reactor Vessel Working Group (RVWG) integrated Generic Letter response.

2.2 Part (2) to NRC Generic Letter 92-01, Revision 1, Supplement 1

Part (2) of the Generic Letter requires the Addressees to provide the following information:

> "an assessment of any change in best-estimate chemistry based on consideration of relevant data;"

To develop and/or verify the best-estimate chemical contents for the North Anna Units 1 and 2 beltline materials and the Surry Units 1 and 2 beltline Rotterdam weld materials, confirmatory searches were performed that included the following sources:

- Westinghouse Electric Company records
- Virginia Power internal documentation
- Industry data bases (i.e., PREP3, PR-EDB, RVID)
- Plant-specific reactor vessel materials surveillance reports

The results of these searches are listed in Tables 2-1 through 2-6. The best-estimate mean copper and nickel chemical contents and their respective standard deviations for each beltline material are also presented in these tables.

The Data Summary Tables for Pressurized Thermal Shock and Upper-Shelf Energy presented in documents BAW-2224[1] and BAW-2222[2] for the North Anna Units 1 and 2 beltline materials and the Surry Units 1 and 2 Rotterdam weld metals respectively have been revised to include the best-estimate copper and nickel chemical compositions described above. These revised tables are presented in Appendix A with the modified values shown in nonshaded boxes.

Table 2-1. Chemistry Data for North Anna Unit 1 Beltline Base Metals

Forging 05 (Nozzle Belt Forging)
Heat No. 990286/295213

Source	Matl. Spec. No.	Cu	Ni	Reference
RDM* Analysis	ASTM A 508 Cl. 2	0.16	0.74	Docket 50-338[3] & RDM Analysis

Forging 04 (Intermediate Shell Forging)
Heat No. 990311/298244

Source	Matl. Spec. No.	Cu	Ni	Reference
RDM* Analysis	ASTM A 508 Cl. 2	0.12	0.82	Docket 50-338[3] & RDM Analysis

Forging 03 (Lower Shell Forging)
Heat No. 990400/292332

Source	Matl. Spec. No.	Cu	Ni	Reference
RDM* Analysis	ASTM A 508 Cl. 2	0.15	0.80	Docket 50-338[3] & RDM Analysis
Surveillance Data	ASTM A 508 Cl. 2	**0.16**	**0.79**	WCAP-8771[4]
Surveillance Data (Specimen VT71)	ASTM A 508 Cl. 2	**0.158**	**0.893**	WCAP-11777[5]
	Mean:	0.16	0.83	
	Std. Dev.:	0.004	0.046	

* - RDM = Rotterdam Dockyard Company.

Table 2-2. Chemistry Data for North Anna Unit 1 Beltline Welds

Weld Wire 25295, Type SMIT 40
Weld 05A - Nozzle Belt to Intermediate Shell Circ. Weld

Source	Flux		Cu	Ni	Reference
	Type	Lot No.			
RDM* Weld Qualification	SMIT 89	1170	0.30	-----	Docket 50-338[3] & Weld Qualification
RDM* Weld Qualification	SMIT 89	1170	0.46	-----	Weld Qualification
RDM* Weld Qualification	SMIT 89	1135	0.25	-----	Weld Qualification**
Surveillance Data	SMIT 89	2275	**0.33**	**0.17**	WCAP-8233[6]
Surveillance Data (Specimen TW58)	SMIT 89	2275	**0.42**	**0.08**	WCAP-10340[7]
		Mean:	0.35	0.13	
		Std. Dev.:	0.077	0.045	

Weld Wire 4278, Type S4Mo
Weld 05B - Nozzle Belt to Intermediate Shell Circ. Weld

Source	Flux		Cu	Ni	Reference
	Type	Lot No.			
RDM* Weld Qualification	SMIT 89	1211	0.11	-----	Docket 50-338[3] & Weld Qualification
Surveillance Data	SMIT 89	1211	**0.13**	**0.11**	WCAP-8513[8]
		Mean:	0.12	0.11	
		Std. Dev.:	0.010	-----	

Weld Wire 25531, Type SMIT 40
Weld 04 - Intermediate Shell to Lower Shell Circ. Weld

Source	Flux		Cu	Ni	Reference
	Type	Lot No.			
Surveillance Data	SMIT 89	1211	**0.086**	**0.11**	WCAP-8771[4]
Surveillance Data (Specimen VW71)	SMIT 89	1211	**0.124**	**0.152**	WCAP-11777[5]
		Mean:	0.11	0.13	
		Std. Dev.:	0.019	0.021	

* - RDM = Rotterdam Dockyard Company.

** - Data obtained from RPVDATA, Reactor Vessel Materials Database, Version 1.0[9]

Table 2-3. Chemistry Data for North Anna Unit 2 Beltline Base Metals

Forging 05 (Nozzle Belt Forging)
Heat No. 990598/291396

Source	Matl. Spec. No.	Cu	Ni	Reference
RDM* Analysis	ASTM A 508 Cl. 2	0.08	0.77	Docket 50-338[3] & RDM Analysis

Forging 04 (Intermediate Shell Forging)
Heat No. 990496/292424

Source	Matl. Spec. No.	Cu	Ni	Reference
RDM* Analysis	ASTM A 508 Cl. 2	0.09	0.83	Docket 50-338[3] & RDM Analysis
Surveillance Data	ASTM A 508 Cl. 2	**0.11**	**0.86**	WCAP-8772[10]
	Mean:	0.10	0.85	
	Std. Dev.:	0.010	0.015	

Forging 03 (Lower Shell Forging)
Heat No. 990533/297355

Source	Matl. Spec. No.	Cu	Ni	Reference
RDM* Analysis	ASTM A 508 Cl. 2	0.13	0.83	Docket 50-388[3] & RDM Analysis

* - RDM = Rotterdam Dockyard Company.

Table 2-4. Chemistry Data for North Anna Unit 2 Beltline Welds

Weld Wire 4278, Type S4Mo
Weld 05A - Nozzle Belt to Intermediate Shell Circ. Weld

Source	Flux Type	Flux Lot No.	Cu	Ni	Reference
RDM* Weld Qualification	SMIT 89	1211	0.11	-----	Docket 50-338[3] & Weld Qualification
Surveillance Data	SMIT 89	1211	**0.13**	**0.11**	WCAP-8513[8]
		Mean:	0.12	0.11	
		Std. Dev.:	0.010	-----	

Weld Wire 801, Type S4Mo
Weld 05B - Nozzle Belt to Intermediate Shell Circ. Weld

Source	Flux Type	Flux Lot No.	Cu	Ni	Reference
RDM* Weld Qualification	SMIT 89	1211	0.18	0.11**	Weld Qualification

Weld Wire 716126, Type S3Mo
Weld 04 - Intermediate Shell to Lower Shell Circ. Weld

Source	Flux Type	Flux Lot No.	Cu	Ni	Reference
RDM* Weld Qualification	LW320	26	0.064	0.04	Weld Qualification***
RDM* Weld Qualification	LW320	26	0.062	0.08	Weld Qualification***
RDM* Weld Qualification	LW320	26	0.079	0.04	Weld Qualification***
RDM* Weld Qualification	LW320	26	0.061	0.03	Weld Qualification***
RDM* Weld Qualification	LW320	26	0.062	0.03	Weld Qualification***
Surveillance Data	LW320	26	**0.088**	**0.084**	WCAP-8772[10]
		Mean:	0.07	0.05	
		Std. Dev.:	0.010	0.023	

* - RDM = Rotterdam Dockyard Company.

** - Conservative estimate (mean plus one standard deviation) determined using data from other plants with similar materials to the beltline material, i.e., data from reactor vessels fabricated to the same material specification in the same shop and in the same time period.[1]

*** - Data obtained from RPVDATA, Reactor Vessel Materials Database, Version 1.0[9]

Table 2-5. Chemistry Data for Surry Unit 1 Rotterdam Weld

Weld Wire 25017, Type SMIT 40 (J726)
Nozzle Belt to Intermediate Shell Circ. Weld

Source	Flux		Cu	Ni	Reference
	Type	Lot No.			
RDM* Weld Qualification	SMIT 89	1197	0.33	0.10**	Docket 50-281[11] & Weld Qualification***

* - RDM = Rotterdam Dockyard Company.

** - Conservative estimate (mean plus one standard deviation) determined using data from other plants with similar materials to the beltline material, i.e., data from reactor vessels fabricated to the same material specification in the same shop and in the same time period.[2]

*** - Data obtained from RPVDATA, Reactor Vessel Materials Database, Version 1.0[9]

Table 2-6. Chemistry Data for Surry Unit 2 Rotterdam Welds

Weld Wire 4275, Type S4Mo (L737)
Nozzle Belt to Intermediate Shell Circ. Weld

Source	Flux		Cu	Ni	Reference
	Type	Lot No.			
Estimated Values	SMIT 89	2275	0.35	0.10*	Docket 50-281[11]

Weld Wire 0227, Type S3Mo (R3008)
Intermediate Shell to Lower Shell Circ. Weld

Source	Flux		Cu	Ni	Reference
	Type	Lot No.			
Surveillance Data	Grau Lo	LW320	**0.19**	**0.56**	WCAP-8085[12]
Surveillance Data (Specimen W14)	Grau Lo	LW320	**0.184**	**0.53**	WCAP-11499[13]
		Mean:	0.19	0.55	
		Std. Dev.:	0.003	0.015	

* - Conservative estimate (mean plus one standard deviation) determined using data from other plants with similar materials to the beltline material, i.e., data from reactor vessels fabricated to the same material specification in the same shop and in the same time period.[2]

2.3 Part (3) to NRC Generic Letter 92-01, Revision 1, Supplement 1

Part (3) of the Generic Letter requires the Addressees to provide the following information:

> "a determination of the need for use of the ratio procedure in accordance with the established Position 2.1 of Regulatory Guide 1.99, Revision 2, for those licensees that use surveillance data to provide a basis for the RPV integrity evaluation;"

If the mean copper and nickel concentrations of surveillance materials are less than the mean copper and nickel concentrations of the beltline materials which they are intended to represent, the use of the surveillance data to calculate ΔRT_{NDT} may under-predict the embrittlement of the reactor vessel beltline. Therefore, Regulatory Guide 1.99, Revision 2,[14] requires measured values of ΔRT_{NDT} to be adjusted when there is clear evidence that the mean copper and nickel concentrations of the surveillance material differ from those of the vessel material. Position 2.1 of Regulatory Guide 1.99, Revision 2, specifies that measured values of ΔRT_{NDT} should be multiplied by the ratio of the chemistry factor for the vessel material (determined in accordance with Position 1.1 of Regulatory Guide 1.99, Revision 2) to that of the surveillance material (also determined in accordance with Regulatory Guide 1.99, Revision 2, Position 1.1). When the mean copper and nickel concentrations of the surveillance material are greater than the mean copper and nickel concentrations of the beltline material, it is not necessary to apply this ratio procedure, since its application would result in less limiting values of ΔRT_{NDT} for the beltline material.

The following is a list of the beltline materials for North Anna Units 1 and 2 and Rotterdam weld materials for Surry Units 1 and 2 that have used surveillance data in evaluations of reactor vessel integrity:

Plant	Heat No.	Description
North Anna Unit 1	990400/292332 25531	Lower Shell Forging 03 Interm. to Lower Shell Circ. Weld
North Anna Unit 2	990496/292424 716126	Interm. Shell Forging 04 Interm. to Lower Shell Circ. Weld
Surry Unit 1	-----	-----
Surry Unit 2	0227	Interm. to Lower Shell Circ. Weld

An examination of the chemical composition data which substantiate the mean values for the beltline materials reveals that the mean copper and nickel concentrations for the surveillance materials are equal to or greater than the mean copper and nickel concentrations determined herein for the corresponding beltline materials. (Values used in the determination of mean copper and nickel concentrations for surveillance materials are presented in **bold** in Tables 2-1 through 2-6). Therefore, it may be concluded that prior uses of surveillance data in evaluations of reactor vessel integrity are conservative with or without application of the Regulatory Guide 1.99, Revision 2, Position 2.1, ratio procedure.

2.4 Part (4) to NRC Generic Letter 92-01, Revision 1, Supplement 1

Part (4) of the Generic Letter requires the Addressees to provide the following information:

> "a written report providing any newly acquired data as specified above and (1) the results of any necessary revisions to the evaluation of RPV integrity in accordance with the requirements of 10CFR50.60, 10CFR50.61, Appendices G and H to 10CFR Part 50, and any potential impact on the LTOP or P-T limits in the technical specifications or (2) a certification that previously submitted evaluations remain valid. Revised evaluations should include consideration of Position 2.1 of Regulatory Guide 1.99, Revision 2, as applicable, and any new data."

Best-estimate (mean) copper and nickel concentrations were previously provided in the Generic Letter 92-01, Revision 1 responses for North Anna Units 1 and 2[1] and Surry Units 1 and 2.[2] After consideration of all relevant data, the mean copper and nickel concentrations previously provided for the North Anna Unit 1 and 2 beltline materials and the Surry Unit 2 beltline Rotterdam weld metal are being updated as follows:

North Anna Unit 1

Material	Previous Copper Conc. (wt%)	Previous Nickel Conc. (wt%)	New Copper Conc. (wt%)	New Nickel Conc. (wt%)
Lower Shell Forging (03)	0.15	0.80	0.16	0.83
Nozzle to Int. Shell Circ. Weld 05A	0.30	0.17	0.35	0.13
Nozzle to Int. Shell Circ. Weld 05B	0.11	0.11	0.12	0.11
Int. to Lower Shell Circ. Weld 04	0.09	0.11	0.11	0.13

North Anna Unit 2

Material	Previous Copper Conc. (wt%)	Previous Nickel Conc. (wt%)	New Copper Conc. (wt%)	New Nickel Conc. (wt%)
Intermediate Shell Forging (04)	0.09	0.83	0.10	0.85
Nozzle to Int. Shell Circ. Weld 05A	0.11	0.11	0.12	0.11
Int. to Lower Shell Circ. Weld 04	0.09	0.08	0.07	0.05

Surry Unit 2

Material	Previous Copper Conc. (wt%)	Previous Nickel Conc. (wt%)	New Copper Conc. (wt%)	New Nickel Conc. (wt%)
Int. to Lower Shell Circ. R3008	0.19	0.56	0.19	0.55

There are no changes to previously reported mean copper and nickel concentrations for Surry Unit 1 Rotterdam beltline weld metal.

Pressurized Thermal Shock Evaluations

As a result of the data collected, the mean copper and nickel concentrations for the North Anna Unit 1 lower shell Forging 03, nozzle belt/intermediate shell circumferential Welds 05A and 05B, and the intermediate/lower shell circumferential Weld 04 beltline materials have increased. Despite the chemical composition changes, the lower shell Forging 03 remains the limiting material with respect to pressurized thermal shock events North Anna Unit 1. (The limiting material determination is based on the relative margin between the 10CFR50.61[15] pressurized thermal shock reference temperature (RT_{PTS}) and the applicable screening criterion.) After consideration of all relevant chemical composition data, the end-of-license RT_{PTS} value for North Anna Unit 1 lower shell Forging 03 increased from 227.7°F to 238.9°F, versus a screening criterion of 270°F. (This value is based on an end-of-license fluence of 3.95E+19 n/cm^2, an initial RT_{NDT} of 38°F, a margin term of 34°F, and the copper and nickel chemical composition described above.) Surveillance data for Forging 03 indicates that the amount of embrittlement is less than the 10CFR50.61 projected value. (When surveillance data is considered, the applicable chemistry factor is reduced from 115°F to 73.5°F.) It is concluded that the chemical composition changes determined herein do not cause the calculated RT_{PTS} value for any North Anna Unit 1 beltline material to exceed the PTS screening criteria.

Similarly, the mean copper and nickel concentrations for the North Anna Unit 2 intermediate shell Forging 04 and nozzle/intermediate shell circumferential Weld 05A have increased, while the mean copper and nickel concentrations for the intermediate/lower shell circumferential Weld

04 have decreased. Despite the chemical composition changes, the lower shell Forging 03 remains the limiting material for North Anna Unit 2 with respect to PTS. Because no new data is currently available for North Anna Unit 2 Forging 03, the PTS evaluation for North Anna Unit 2 is conservative and remains valid.

For Surry Unit 1, the limiting material with respect to PTS is the B&W-fabricated Linde 80 weld metal, SA-1585. The evaluation of Surry Units 1 and 2 Linde 80 weld materials is presented under separate cover.[16] For Surry Unit 2 the limiting material with respect to PTS is the intermediate/lower shell circumferential Rotterdam Weld R3008. Because the mean copper concentration for this weld is unchanged, and the mean nickel concentration is less than that previously reported,[2] it is concluded that the current PTS evaluation for Surry Unit 2 is conservative and remains valid.

Low Temperature Overpressure Protection System (LTOPS) and Pressure-Temperature (P-T) Limit Evaluations

The current LTOPS and P-T limit evaluations for North Anna Unit 1 are based on the limiting weld identified as Weld 04 (intermediate/lower shell circumferential weld). The adjusted reference temperature (ART) for the applicable time period was calculated on the basis of surveillance data. Since the recorded copper and nickel concentrations for the surveillance material are greater than the best-estimate copper and nickel concentrations for the beltline material, the current LTOPS and P-T limit evaluations for North Anna Unit 1 are conservative and remain valid.

The current LTOPS and P-T limit evaluations for North Anna Unit 2 are based on the limiting material identified as Forging 03 (Lower Shell Forging). The ART for the applicable time period was calculated in accordance with the Regulatory Guide 1.99, Revision 2, Position 1.1 guidelines. Since no new data is currently available for this limiting material, the current LTOPS and P-T limit evaluation for North Anna Unit 2 remain valid.

Recently submitted LTOPS and P-T evaluations for Surry Units 1 and 2 were based on the Surry Unit 1 intermediate/lower shell circumferential weld SA-1585,[17] which is a Linde 80 weld material. This material remains the limiting material for both Surry Units with respect to LTOPS and P-T limit evaluations. The response to Part (4) of NRC Generic Letter 92-01, Revision 1, Supplement 1, for the B&W Owners Group (B&WOG) Reactor Vessel Working Group (RVWG) Linde 80 weld metals is included in the B&WOG RVWG integrated response.[16]

Upper-Shelf Energy Evaluations

The response to the NRC closure letter for Generic Letter 92-01, Revision 1, for the North Anna Unit 1 and 2 reactor vessel beltline materials[2] demonstrated compliance with the upper-shelf energy (USE) requirements of 10CFR50 Appendix G.[18] Regulatory Guide 1.99, Revision 2 models the percent drop in USE as a function of fluence and mean copper content only. As described above, the mean copper concentrations for several North Anna beltline materials have increased. However, after consideration of the revised copper concentrations, the T/4 end-of-license USE values for the North Anna Units 1 and 2 beltline materials remain above the 50 ft-lb level.

The limiting materials with respect to USE for Surry Unit 1 and 2 are the Linde 80 weld metals. Because the mean copper concentrations for the Rotterdam weld metals in the Surry Units 1 and 2 reactor vessels beltline remain unchanged from those previously reported, the Linde 80 weld metals remain the limiting materials with respect to USE. The equivalent margin analyses for these weld materials as documented in Topical Reports BAW-2178PA[19] and BAW-2192PA[20] remain valid.

3.0 REFERENCES

1. M. J. DeVan, "North Anna Units 1 and 2 Response to Closure Letter for NRC Generic Letter 92-01, Revision 1," BAW-2224, B&W Nuclear Technologies, Inc., Lynchburg, Virginia, July 1994.

2. M. J. DeVan and K. K. Yoon, "Response to Closure Letters to Generic Letter 92-01, Revision 1," BAW-2222, B&W Nuclear Technologies, Inc., Lynchburg, Virginia, June 1994.

3. Letter from C. M. Stallings, Virginia Electric and Power Company to Harold B. Denton, Office of Nuclear Reactor Regulation, "Pressure Vessel Fracture Toughness Properties, North Anna Power Station, Unit Nos. 1 and 2," Docket No. 50-338, December 11, 1978, Public Document Accession No. 7812150277.

4. J. A. Davidson and J. H. Phillips, "Virginia Electric and Power Company North Anna Unit No. 1 Reactor Vessel Radiation Surveillance Program," WCAP-8771, Westinghouse Electric Corporation, Pittsburgh, Pennsylvania, September 1976.

5. S. E. Yanichko, et al., "Analysis of Capsule U from the Virginia Electric and Power Company North Anna Unit 1 Reactor Vessel Radiation Surveillance Program," WCAP-11777, Westinghouse Electric Corporation, Pittsburgh, Pennsylvania, February 1988.

6. S. E. Yanichko, D. J. Lege, and J. H. Phillips, "Tennessee Valley Authority Sequoyah Unit No. 1 Reactor Vessel Radiation Surveillance Program," WCAP-8233, Westinghouse Electric Corporation, Pittsburgh, Pennsylvania, November 1973.

7. S. E. Yanichko, et al., Analysis of Capsule T from the Tennessee Valley Authority Sequoyah Unit 1 Reactor Vessel Radiation Surveillance Program," WCAP-10340, Westinghouse Electric Corporation, Pittsburgh, Pennsylvania, May 1983.

8. J. A. Davidson, J. H. Phillips, and S. E. Yanichko, "Tennessee Valley Authority Sequoyah Unit No. 2 Reactor Vessel Radiation Surveillance Program," WCAP-8513, Westinghouse Electric Corporation, Pittsburgh, Pennsylvania, November 1975.

9. RPVDATA, Reactor Vessel Materials Database, Version 1.0, developed for Westinghouse Owners Group by ATI Consulting, dated July 28, 1995.

10. J. A. Davidson, J. H. Phillips, and S. E. Yanichko, "Virginia Electric and Power Company North Anna Unit No. 2 Reactor Vessel Radiation Surveillance Program," WCAP-8772, Westinghouse Electric Corporation, Pittsburgh, Pennsylvania, November 1976.

11. Letter from C. M. Stallings, Virginia Electric and Power Company to E. G. Case, Office of Nuclear Reactor Regulation, "Reactor Vessel Material Surveillance Program," Docket No. 50-281, January 23, 1978, Public Document Accession No. 780260154.

12. S. E. Yanichko and D. J. Lege, "Virginia Electric and Power Company Surry Unit No. 2 Reactor Vessel Radiation Surveillance Program, WCAP-8085, Westinghouse Electric Corporation, Pittsburgh, Pennsylvania, June 1973.

13. S. E. Yanichko and V. A. Perone, "Analysis of Capsule V from the Virginia Electric Power Company Surry Unit 2 Reactor Vessel Radiation Surveillance Program," WCAP-11499, Westinghouse Electric Corporation, Pittsburgh, Pennsylvania, June 1987.

14. U. S. Nuclear Regulatory Commission, "Radiation Embrittlement of Reactor Vessel Materials," Regulatory Guide 1.99, Revision 2, May 1988.

15. Code of Federal Regulations, Title 10, Part 50.61, "Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events."

16. M. J. DeVan, "B&W Owners Group Reactor Vessel Working Group Response to Generic Letter 92-01, Revision 1, Supplement 1," BAW-2257, Revision 1, B&W Nuclear Technologies, Inc., Lynchburg, Virginia, October 1995.

17. M. J. Malone, "Surry Units 1 and 2 Heatup and Cooldown Limit Curves for Normal Operation," WCAP-14177, Westinghouse Electric Corporation, Pittsburgh, Pennsylvania, October 1994.

18. Code of Federal Regulations, Title 10, Part 50, "Domestic Licensing of Production and Utilization Facilities," Appendix G, Fracture Toughness Requirements.

19. K. K. Yoon, "Low Upper-Shelf Toughness Fracture Analysis of Reactor Vessels of B&W Owners Group Reactor Vessel Working Group for Level C & D Service Loads," BAW-2178PA, B&W Nuclear Technologies, Inc., Lynchburg, Virginia, April 1994.

20. K. K. Yoon, "Low Upper-Shelf Toughness Fracture Analysis of Reactor Vessels of B&W Owners Group Reactor Vessel Working Group for Load Level A & B Conditions," BAW-2192PA, B&W Nuclear Technologies Inc., Lynchburg, Virginia, April 1994.

4.0 CERTIFICATION

This report accurately responds to the information requested in Generic Letter 92-01, Revision 1, Supplement 1.

10/31/95

M. J. DeVan, Engineer III — Date
Materials & Structural Analysis Unit

This report has been reviewed for technical content and accuracy.

Oct. 31, 1995

L. B. Gross, Advisory Engineer — Date
Materials & Structural Analysis Unit

Verification of independent review.

10-31-95

K. E. Moore, Manager — Date
Materials & Structural Analysis Unit

This report is approved for release.

10/31/95

D. L. Howell — Date
Program Manager

APPENDIX A

Revised Pressurized Thermal Shock and Upper-Shelf Energy Data Summary Tables

Table A-1. North Anna Unit 1 -- Data Summary for Pressurized Thermal Shock Calculation

Beltline Material	Heat No.	IS Neutron Fluence at 30.7 EFPY	IRT_{NDT} F	Method of Determin. IRT_{NDT}	Chemistry Factor	Method of Determin. CF	%Cu	%Ni
Nozzle Belt Shell Forging 05	990286/ 295213	2.77E+18[(a)]	+6	MTEB 5-2[(b)]	121.5	RG1.99 Table 2[(e)]	0.16[(g)]	0.74[(g)]
Interm. Shell Forging 04	990311/ 298244	3.95E+19[(a)]	+17	Plant Specific[(c)]	86	RG1.99 Table 2[(e)]	0.12[(g)]	0.82[(g)]
Lower Shell Forging 03	990400/ 292332	3.95E+19[(a)]	+38	Plant Specific[(c)]	88.9	Calculated[(f)]	0.16[(g)]	0.83[(g)]
Weld 05A NB to IS Circ. Weld (OD 94%)	25295	2.77E+18[(a)]	0	Generic[(d)]	162.75	RG1.99 Table 1[(e)]	0.35[(g)]	0.13[(g)]
Weld 05B NB to IS Circ. Weld (ID 6%)	4278	2.77E+18[(a)]	0	Generic[(d)]	63.0	RG1.99 Table 1[(e)]	0.12[(g)]	0.11[(g)]
Weld 04 IS to LS Circ. Weld	25531	3.95E+19[(a)]	+19	Plant Specific[(c)]	93.1	Calculated[(f)]	0.11[(g)]	0.13[(g)]

Table A-1. (cont.) North Anna Unit 1 -- Data Summary for Pressurized Thermal Shock Calculation

NOTES:

a. Values obtained from WCAP-11777.[A-1] (Nozzle belt shell forging and nozzle belt shell-to-intermediate shell circumferential weld fluences are 7% of maximum vessel inner surface fluence.)

b. Initial reference temperature was determined in accordance with MTEB 5-2[A-2] guidelines for cases where the reference temperature was not determined using ASME Boiler and Pressure Vessel Code, Section III, NB-2331,[A-3] methodology.

c. Initial reference temperature was determined from tests on material fabricated from the same heat of the beltline material.

d. Initial reference temperature was determined from the mean value of tests on material of similar types.

e. Chemistry factor was determined from the chemistry factor tables in Regulatory Guide 1.99, Revision 2.[A-4]

f. Chemistry factor was determined from surveillance data (WCAP-11777) via procedures described in Regulatory Guide 1.99, Revision 2.

g. Best-estimate values.

Table A-2. North Anna Unit 1 -- Data Summary for Upper-Shelf Energy Calculation

Beltline Material	Heat No.	Material Type	1/4T USE at 30.7 EFPY	1/4T Neutron Fluence at 30.7 EFPY	Unirrad. USE	Method of Determin. Unirrad. USE
Nozzle Belt Shell Forging 05	990286/ 295213	A 508-2	62	1.75E+18[(a)]	74	Calculated[(b)]
Interm. Shell Forging 04	990311/ 298244	A 508-2	68	2.49E+19[(a)]	92	Direct[(c)]
Lower Shell Forging 03	990400/ 292332	A 508-2	59	2.49E+19[(a)]	85	Direct[(c)]
Weld 05A NB to IS Circ. Weld (OD 94%)	25295	SMIT 89, SAW	75	1.75E+18[(a)]	111	Sister Plant[(d)]
Weld 05B NB to IS Circ. Weld (ID 6%)	4278	SMIT 89, SAW	---[(e)]	1.75E+18[(a)]	105	Sister Plant[(d)]
Weld 04 IS to LS Circ. Weld	25531	SMIT 89, SAW	70	2.49E+19[(a)]	102	Direct[(c)]

Table A-2. (cont.) North Anna Unit 1 -- Data Summary for Upper-Shelf Energy Calculation

NOTES:

a. End-of-life neutron fluence at T/4 from inner wall calculated using Regulatory Guide 1.99, Revision 2,[A-4] neutron fluence attenuation methodology from ID value. (Vessel thickness = 7.667 in.)

b. The unirradiated USE was determined on the basis of a comparison with similar materials to the beltline material.

c. The unirradiated USE for the forgings was determined from weak oriented specimens. The unirradiated USE for the weld was determined from test data.

d. The unirradiated USE was determined using reported data from other plants with the same weld wire heat number (Sequoyah Units 1 and 2[A-5,A-6]).

e. Weld 05A is 94% of the thickness of the nozzle belt shell-to-intermediate shell circumferential weld and Weld 05B is the remainder. Therefore, it is not necessary to evaluate the end-of-life USE for Weld 05B because it is not at the T/4 location.

Table A-3. North Anna Unit 2 -- Data Summary for Pressurized Thermal Shock Calculation

Beltline Material	Heat No.	IS Neutron Fluence at 32 EFPY	IRT_{NDT} F	Method of Determin. IRT_{NDT}	Chemistry Factor	Method of Determin. CF	%Cu	%Ni
Nozzle Belt Shell Forging 05	990598/ 291396	3.13E+18[(a)]	+9	MTEB 5-2[(b)]	51	RG1.99 Table 2[(e)]	0.08[(g)]	0.77[(g)]
Interm. Shell Forging 04	990496/ 292424	4.47E+19[(a)]	+75	Plant Specific[(c)]	47.9	Calculated[(f)]	0.10[(g)]	0.85[(g)]
Lower Shell Forging 03	990533/ 297355	4.47E+19[(a)]	+56	Plant Specific[(c)]	96	RG1.99 Table 2[(e)]	0.13[(g)]	0.83[(g)]
Weld 05A NB to IS Circ. Weld (OD 94%)	4278	3.13E+18[(a)]	0	Generic[(d)]	63.0	RG1.99 Table 1[(e)]	0.12[(g)]	0.11[(g)]
Weld 05B NB to IS Circ. Weld (ID 6%)	801	3.13E+18[(a)]	0	Generic[(d)]	87.8	RG1.99 Table 1[(e)]	0.18[(g)]	0.11[(g)]
Weld 04 IS to LS Circ. Weld	716126	4.47E+19[(a)]	-48	Plant Specific[(c)]	10.4	Calculated[(f)]	0.07[(g)]	0.05[(g)]

Table A-3. (cont.) North Anna Unit 2 -- Data Summary for Pressurized Thermal Shock Calculation

NOTES:

a. Values obtained from WCAP-12497.[A-7] (Nozzle belt shell forging and nozzle belt shell-to-intermediate shell circumferential weld fluences are 7% of maximum vessel inner surface fluence.)

b. Initial reference temperature was determined in accordance with MTEB 5-2[A-2] guidelines for cases where the reference temperature was not determined using ASME Boiler and Pressure Vessel Code, Section III, NB-2331,[A-3] methodology.

c. Initial reference temperature was determined from tests on material fabricated from the same heat of the beltline material.

d. Initial reference temperature was determined from the mean value of tests on material of similar types.

e. Chemistry factor was determined from the chemistry factor tables in Regulatory Guide 1.99, Revision 2.[A-4]

f. Chemistry factor was determined from surveillance data (WCAP-12497) via procedures described in Regulatory Guide 1.99, Revision 2.

g. Best-estimate values.

Table A-4. North Anna Unit 2 -- Data Summary for Upper-Shelf Energy Calculation

Beltline Material	Heat No.	Material Type	1/4T USE at 32 EFPY	1/4T Neutron Fluence at 32 EFPY	Unirrad. USE	Method of Determin. Unirrad. USE
Nozzle Belt Shell Forging 05	990598/ 291396	A 508-2	64	1.98E+18[a]	74	Equiv. to Forging 04[b]
Interm. Shell Forging 04	990496/ 292424	A 508-2	56	2.82E+19[a]	74	Direct[c]
Lower Shell Forging 03	990533/ 297355	A 508-2	58	2.82E+19[a]	80	Direct[c]
Weld 05A NB to IS Circ. Weld (OD 94%)	4278	SMIT 89, SAW	86	1.98E+18[a]	105	Sister Plant[d]
Weld 05B NB to IS Circ. Weld (ID 6%)	801	SMIT 89, SAW	---[e]	1.98E+18[a]	---[e]	---[e]
Weld 04 IS to LS Circ. Weld	716126	LW 320, SAW	78	2.82E+19[a]	107	Direct[c]

Table A-4. (cont.) North Anna Unit 2 -- Data Summary for Upper-Shelf Energy Calculation

NOTES:

a. End-of-life neutron fluence at T/4 from inner wall calculated using Regulatory Guide 1.99, Revision 2,[A-4] neutron fluence attenuation methodology from ID value. (Vessel thickness = 7.667 in.)

b. Letter from W. L. Stewart, Virginia Electric and Power Company, to U. S. Nuclear Regulatory Commission, Subject: Virginia Electric and Power Company North Anna Power Station Unit 2 Selection of Limiting Forged Material for Low Upper-Shelf Energy Considerations, dated December 29, 1992.[A-8]

c. The unirradiated USE for the forgings was determined from weak oriented specimens. The unirradiated USE for the weld was determined from test data.

d. The unirradiated USE was determined using reported data from other plants with the same weld wire heat number (Sequoyah Unit 2[A-5]).

e. Weld 05A is 94% of the thickness of the nozzle belt shell-to-intermediate shell circumferential weld and Weld 05B is the remainder. Therefore, it is not necessary to evaluate the end-of-life USE for Weld 05B because it is not at the T/4 location.

Table A-5. Surry Unit 1 -- Data Summary for Pressurized Thermal Shock Calculation

Beltline Material	Heat No.	IS Neutron Fluence at 28.8 EFPY	IRT_{NDT} F	Method of Determin. IRT_{NDT}	Chemistry Factor	Method of Determin. CF	%Cu	%Ni
Upper Shell Forging	122V109VA1	3.96E+18[a]	+40[b] (σ_I=0)	Plant Specific	58	RG1.99 Table 2	0.09[b]	0.74[b]
Interm. Shell Plate	C4326-1	3.96E19[a]	+10[b] (σ_I=0)	Plant Specific	73.5	RG1.99 Table 2	0.11[h]	0.55[h]
Interm. Shell Plate	C4326-2	3.96E+19[a]	0[b] (σ_I=0)	Plant Specific	73.5	RG1.99 Table 2	0.11[h]	0.55[h]
Lower Shell Plate	C4415-1	3.96E+19[a]	+20[b] (σ_I=0)	Plant Specific	89.2[f]	Calculated	0.11[h]	0.50[h]
Lower Shell Plate	C4415-2	3.96E+19[a]	0[b] (σ_I=0)	Plant Specific	73	RG1.99 Table 2	0.11[h]	0.50[h]
J726 US to IS Circ. Weld	25017	3.96E+18[a]	0[c] (σ_I=20)	Generic	152	RG1.99 Table 1	0.33[i]	0.10[i]
SA-1585 IS to LS Circ. Weld (ID 40%)	72445	3.96E+19[a]	-5[d] (σ_I=19.7)	Generic	149.2[g]	Calculated	0.21[j]	0.59[j]
SA-1650 IS to LS Circ. Weld (OD 60%)	72445	3.96E+19[a]	-5[d] (σ_I=19.7)	Generic	149.2[g]	Calculated	0.21[j]	0.59[j]
SA-1494 IS Axial Welds (L3 - 100%) (L4 - 100%)	8T1554	6.39E+18[a]	-5[d] (σ_I=19.7)	Generic	159.0	RG1.99 Table 1	0.18[j]	0.63[j]

Beltline Material	Heat No.	IS Neutron Fluence at 28.8 EFPY	IRT_{NDT} F	Method of Determin. IRT_{NDT}	Chemistry Factor	Method of Determin. CF	%Cu	%Ni
SA-1494 LS Axial Weld (L1 - 100%)	8T1554	6.39E+18[a]	-5[d] (σ_I=19.7)	Generic	159.0	RG1.99 Table 1	0.18[j]	0.63[j]
SA-1526 LS Axial Weld (L2 - 100%)	299L44	6.39E+18[a]	-7[e] (σ_I=20.6)	Generic	219.4[e]	Calculated	0.35[j]	0.68[j]

Table A-5. (cont.) Surry Unit 1 -- Data Summary for Pressurized Thermal Shock Calculation

NOTES:

a. Values obtained from WCAP-11015, Revision 1,[A-9] or WCAP-11017, Revision 1.[A-10] (Upper shell forging and intermediate-to-upper shell circumferential weld fluences are 10% of maximum vessel inner surface fluence.)

b. Values determined from data in Material Test Report.

c. Since the initial RT_{NDT} data for the Rotterdam welds are similar to the Linde 80 class welds, the initial RT_{NDT} for the Rotterdam welds are bounded by Linde 80 weld metal generic values. Therefore, IRT_{NDT} = 0F, and a σ_I = 20F are estimated for the Rotterdam welds. See SECY 82-465,[A-11] "Pressurized Thermal Shock."

d. Mean value from data in BAW-1803, Revision 1.[A-12]

e. Initial RT_{NDT}, σ_I, and Chemistry Factor for weld metal SA-1526 are based on NRC Safety Evaluation related to Amendment No. 176 to Facility Operating License No. DPR-50, Three Mile Island Nuclear Station, Unit No. 1, Docket No. 50-289.[A-13] Chemistry Factor weld metal WF-25 was determined using TMI1 and B&WOG surveillance data for weld metal WF-25 and S1 surveillance data for weld metal SA-1526. These surveillance welds were fabricated with the same wire heat. The TMI1 and B&WOG surveillance data were obtained from BAW-1803, Revision 1.

f. Chemistry Factor for plate C4415-1 was determined using S1 surveillance data as reported in WCAP-11415.[A-14]

g. Chemistry Factor for weld metal SA-1585 and weld metal SA-1650 was determined using B&WOG surveillance data for weld metal SA-1585 and PB1 surveillance data for weld metal SA-1263. These surveillance welds were fabricated with the same wire heat. The B&WOG surveillance data were obtained from BAW-1803, Revision 1. The PB1 30 ft-lb transition temperature shift data were also obtained from BAW-1803, Revision 1, while the fluence data for the capsules were obtained from WCAP-12794, Revision 2.[A-15]

h. Values obtained from BAW-2150.[A-16]

i. Best-estimate value.

j. Values obtained from BAW-2121P.[A-17]

Table A-6. Surry Unit 1 -- Data Summary for Upper-Shelf Energy Calculation

Beltline Material	Heat No.	Material Type	1/4T USE at 28.8 EFPY	1/4T Neutron Fluence at 28.8 EFPY	Unirrad. USE	Method of Determin. Unirrad. USE
Upper Shell Forging	122V109VA1	A 508-2	72.2	2.64E+18[c]	83	MTEB 5-2[d]: 65% (Matl. Cert.)
Interm. Shell Plate	C4326-1	A 533B-1	87.3	2.20E+19[c]	115	Direct[e] (Surv. Matl.)
Interm. Shell Plate	C4326-2	A 533B-1	71.7	2.20E+19[c]	94	MTEB 5-2[d]: 65% (Matl. Cert.)
Lower Shell Plate	C4415-1	A 533B-1	78.2	2.20E+19[c]	103	Direct[e] (Surv. Matl.)
Lower Shell Plate	C4415-2	A 533B-1	63.0	2.20E+19[c]	83	MTEB 5-2[d]: 65% (Matl. Cert.)
J726 US to IS Circ. Weld	25017	Rotterdam SAW	EMA[a]	2.64E+18[c]	EMA[a]	Note a
SA-1585 IS to LS Circ. Weld (ID 40%)	72445	Linde 80 SAW	EMA[b]	2.20E+19[c]	77 EMA[b]	Direct[f] (Surv. Matl.)
SA-1650 IS to LS Circ. Weld (OD 60%)	72445	Linde 80 SAW	EMA[b]	2.20E+19[c]	77 EMA[b]	Direct[f] (Surv. Matl.)

Beltline Material	Heat No.	Material Type	1/4T USE at 28.8 EFPY	1/4T Neutron Fluence at 28.8 EFPY	Unirrad. USE	Method of Determin. Unirrad. USE
SA-1494 IS Axial Welds (L3 - 100%) (L4 - 100%)	8T1554	Linde 80 SAW	EMA[b]	3.54E+18[c]	EMA[b]	Generic[g]
SA-1494 LS Axial Weld (L1 - 100%)	8T1554	Linde 80 SAW	EMA[b]	3.54E+18[c]	EMA[b]	Generic[g]
SA-1526 LS Axial Weld (L2 - 100%)	299L44	Linde 80 SAW	EMA[b]	3.54E+18[c]	70 EMA[b]	Direct[f] (Surv. Matl.)

Table A-6. (cont.) Surry Unit 1 -- Data Summary for Upper-Shelf Energy Calculation

NOTES:

a. Rotterdam welds were considered in the equivalent margin analyses documented in Topical Reports BAW-2178PA[A-18] and BAW-2192PA[A-19] and found to be acceptable by the NRC, as stated in the applicable Safety Evaluation Reports.

b. The approved equivalent margin analyses in the Topical Reports BAW-2192PA and BAW-2178P demonstrates compliance with requirements of 10CFR50, Appendix G.[A-20]

c. Values obtained from BAW-2192A, based on a conservative estimated wall thickness of 7.75 inches. (Rotterdam fabrication report indicates that the minimum vessel wall thickness is 8.0 inches.) Fluence estimates do not include the benefit of the installation of flux suppression inserts beginning with Surry Unit 1 Cycle 13. End-of-life extension flux reduction targets for Surry Unit 1 are 2.5E+19 n/cm^2 at 0° 1/4T location and 7.5E+18 n/cm^2 at the 45° inner surface location.

d. Unirradiated USE is 65% of the USE from a longitudinal oriented specimens as defined in MTEB 5-2.[A-2]

e. Unirradiated USE is determined from transverse oriented specimens (WCAP-11415[A-14]).

f. Unirradiated USE is determined from surveillance weld specimens (BAW-1803, Revision 1[A-12]).

g. Unirradiated USE is determined using data from other plants with similar materials to the beltline material.

Table A-7. Surry Unit 2 -- Data Summary for Pressurized Thermal Shock Calculation

Beltline Material	Heat No.	IS Neutron Fluence at 29.4 EFPY	IRT_{NDT} F	Method of Determin. IRT_{NDT}	Chemistry Factor	Method of Determin. CF	%Cu	%Ni
Upper Shell Forging	123V303VA1	3.43E+18[a]	+30[b] (σ_I=0)	Plant Specific	58	RG1.99 Table 2	0.09[b]	0.73[b]
Interm. Shell Plate	C4331-2	3.43E+19[a]	-10[b] (σ_I=0)	Plant Specific	83	RG1.99 Table 2	0.12[h]	0.60[h]
Interm. Shell Plate	C4339-2	3.43E+19[a]	-20[b] (σ_I=0)	Plant Specific	73.4	RG1.99 Table 2	0.11[h]	0.54[h]
Lower Shell Plate	C4208-2	3.43E+19[a]	-30[b] (σ_I=0)	Plant Specific	107.3	RG1.99 Table 2	0.15[h]	0.55[h]
Lower Shell Plate	C4339-1	3.43E+19[a]	-10[b] (σ_I=0)	Plant Specific	68.4[e]	Calculated	0.11[h]	0.54[h]
L737 US to IS Circ. Weld	4275	3.43E+18[a]	0[c] (σ_I=20)	Generic	160.5	RG1.99 Table 1	0.35[i]	0.10[i]
R3008 IS to LS Circ. Weld	0227	3.43E+19[a]	0[c] (σ_I=20)	Generic	128.0[f]	Calculated	0.19[i]	0.55[i]
WF-4 IS Axial Weld (L4 - ID 50%)	8T1762	7.14E+18[a]	-5[d] (σ_I=19.7)	Generic	152.3	RG1.99 Table 1	0.20[i]	0.55[i]

Beltline Material	Heat No.	IS Neutron Fluence at 29.4 EFPY	IRT_{NDT} F	Method of Determin. IRT_{NDT}	Chemistry Factor	Method of Determin. CF	%Cu	%Ni
SA-1585 IS Axial Welds (L4 - OD 50%) (L3 - 100%)	72445	7.14E+18[a]	-5[d] (σ_I=19.7)	Generic	149.2[g]	Calculated	0.21[j]	0.59[j]
WF-4 LS Axial Welds (L1 - 100%) (L2 - ID 63%)	8T1762	7.14E+18[a]	-5[d] (σ_I=19.7)	Generic	152.3	RG1.99 Table 1	0.20[j]	0.55[j]
WF-8 LS Axial Weld (L2 - OD 37%)	8T1762	7.14E+18[a]	-5[d] (σ_I=19.7)	Generic	152.3	RG1.99 Table 1	0.20[j]	0.55[j]

Table A-7. (cont.) Surry Unit 2 -- Data Summary for Pressurized Thermal Shock Calculation

NOTES:

a. Values obtained from WCAP-11015, Revision 1,[A-9] or WCAP-11017, Revision 1.[A-10] (Upper shell forging and intermediate-to-upper shell circumferential weld fluences are 10% of maximum vessel inner surface fluence.)

b. Values determined from data in Material Test Report.

c. Since the initial RT_{NDT} data for the Rotterdam welds are similar to the Linde 80 class welds, the initial RT_{NDT} for the Rotterdam welds are bounded by Linde 80 weld metal generic values. Therefore, IRT_{NDT} = 0F, and a σ_I = 20F are estimated for the Rotterdam welds. See SECY 82-465,[A-11] "Pressurized Thermal Shock."

d. Mean value from data in BAW-1803, Revision 1.[A-12]

e. Chemistry Factor for plate C4339-1 was determined using S2 surveillance data as reported in WCAP-11499.[A-21]

f. Chemistry Factor for weld metal R3008 was determined using S2 surveillance data as reported in WCAP-11499.

g. Chemistry Factor for weld metal SA-1585 was determined using B&WOG surveillance data for weld metal SA-1585 and PB1 surveillance data for weld metal SA-1263. These surveillance welds were fabricated with the same wire heat. The B&WOG surveillance data were obtained from BAW-1803, Revision 1. The PB1 30 ft-lb transition temperature shift data were also obtained from BAW-1803, Revision 1, while the fluence data for the capsules were obtained from WCAP-12794, Revision 2.[A-15]

h. Values obtained from BAW-2150.[A-16]

i. Best-estimate value.

j. Values obtained from BAW-2121P.[A-17]

Table A-8. Surry Unit 2 -- Data Summary for Upper-Shelf Energy Calculation

Beltline Material	Heat No.	Material Type	1/4T USE at 29.4 EFPY	1/4T Neutron Fluence at 29.4 EFPY	Unirrad. USE	Method of Determin. Unirrad. USE
Upper Shell Forging	123V303VA1	A 508-2	90.7	2.23E+18[c]	104	MTEB 5-2[d]: 65% (Matl. Cert.)
Interm. Shell Plate	C4331-2	A 533B-1	63.6	1.86E+19[c]	84	MTEB 5-2[d]: 65% (Matl. Cert.)
Interm. Shell Plate	C4339-2	A 533B-1	63.8	1.86E+19[c]	83	MTEB 5-2[d]: 65% (Matl. Cert.)
Lower Shell Plate	C4208-2	A 533B-1	67.9	1.86E+19[c]	94	MTEB 5-2[d]: 65% (Matl. Cert.)
Lower Shell Plate	C4339-1	A 533B-1	80.7	1.86E+19[c]	105	Direct[e] (Surv. Matl.)
L737 US to LS Circ. Weld	4275	Rotterdam SAW	EMA[a]	2.23E+18[c]	Note a	Note a
R3008 IS to LS Circ. Weld	0227	Rotterdam SAW	56.5 EMA[b]	1.86E+19[c]	90 EMA[a]	Direct[e] (Surv. Matl.)
WF-4 IS Axial Weld (L4 - ID 50%)	8T1762	Linde 80 SAW	EMA[b]	3.88E+18[c]	EMA[b]	Generic[f]

Beltline Material	Heat No.	Material Type	1/4T USE at 29.4 EFPY	1/4T Neutron Fluence at 29.4 EFPY	Unirrad. USE	Method of Determin. Unirrad. USE
SA-1585 IS Axial Welds (L4 - OD 50%) (L3 - 100%)	72445	Linde 80 SAW	EMA[b]	3.88E+18[c]	77 EMA[b]	Direct[g]
WF-4 LS Axial Welds (L1 - 100%) (L2 - ID 63%)	8T1762	Linde 80 SAW	EMA[b]	3.88E+18[c]	EMA[b]	Generic[f]
WF-8 LS Axial Weld (L2 - OD 37%)	8T1762	Linde 80 SAW	EMA[b]	3.88E+18[c]	EMA[b]	Generic[f]

Table A-8. (cont.) Surry Unit 2 -- Data Summary for Upper-Shelf Energy Calculation

NOTES:

a. Rotterdam welds were considered in the equivalent margin analyses documented in Topical Reports BAW-2178PA[A-18] and BAW-2192PA[A-19] and found to be acceptable by the NRC, as stated in the applicable Safety Evaluation Reports.

b. The approved equivalent margin analyses in the Topical Reports BAW-2192PA and BAW-2178P demonstrates compliance with requirements of 10CFR50, Appendix G.[A-20]

c. Values obtained from BAW-2192A, based on a conservative estimated wall thickness of 7.75 inches. (Rotterdam fabrication report indicates that the minimum vessel wall thickness is 8.0 inches.)

d. Unirradiated USE is 65% of the USE from a longitudinal oriented specimens as defined in MTEB 5-2.[A-2]

e. Unirradiated USE is determined from transverse oriented specimens (WCAP-11499[A-21]).

f. Unirradiated USE is determined using data from other plants with similar materials to the beltline material.

g. Unirradiated USE is determined from surveillance weld specimens (BAW-1803, Revision 1[A-12]).

APPENDIX A

References

A-1. S. E. Yanichko, et al., "Analysis of Capsule U from the Virginia Electric and Power Company North Anna Unit 1 Reactor Vessel Radiation Surveillance Program," WCAP-11777, Westinghouse Electric Corporation, Pittsburgh, Pennsylvania, February 1988.

A-2. U. S. Nuclear Regulatory Commission, Standard Review Plan, Branch Technical Position MTEB 5-2, Revision 1, "Fracture Toughness Requirements, NUREG-0800, July 1981.

A-3. American Society of Mechanical Engineers, Boiler and Pressure Vessel Code, Section III, "Nuclear Power Plant Components," NB-2331, 1989 Edition.

A-4. U. S. Nuclear Regulatory Commission, "Radiation Embrittlement of Reactor Vessel Materials," Regulatory Guide 1.99, Revision 2, May 1988.

A-5. J. A. Davidson, J. H. Phillips, and S. E. Yanichko, "Tennessee Valley Authority Sequoyah Unit No. 2 Reactor Vessel Radiation Surveillance Program," WCAP-8513, Westinghouse Electric Corporation, Pittsburgh, Pennsylvania, November 1975.

A-6. S. E. Yanichko, D. J. Lege, and J. H. Phillips, "Tennessee Valley Authority Sequoyah Unit No. 1 Reactor Vessel Radiation Surveillance Program," WCAP-8233, Westinghouse Electric Corporation, Pittsburgh, Pennsylvania, November 1973.

A-7. E. Terek, S. L. Anderson, and L. Albertin "Analysis of Capsule U from the Virginia Electric and Power Company North Anna Unit 2 Reactor Vessel Radiation Surveillance Program," WCAP-12497, Westinghouse Electric Corporation, Pittsburgh, Pennsylvania, January 1990.

A-8. Letter from W. L. Stewart, Virginia Electric and Power Company, to U. S. Nuclear Regulatory Commission, "Virginia Electric and Power Company North Anna Power Station Unit 2 Selection of Limiting Forged Material for Low Upper-Shelf Energy Considerations," December 28, 1992.

A-9. C. C. Heinecke, et al., "Surry Units 1 and 2 Reactor Vessel Fluence and RT_{PTS} Evaluations," WCAP-11015, Revision 1, Westinghouse Electric Corporation, Pittsburgh, Pennsylvania, April 1987.

A-10. C. C. Heinecke, et al., "Surry Units 1 and 2 Reactor Vessel Fluence and RT_{PTS} Evaluations for Consideration of Life Extension," WCAP-11017, Revision 1, Westinghouse Electric Corporation, Pittsburgh, Pennsylvania, April 1987.

A-11. U. S. Nuclear Regulatory Commission, "Pressurized Thermal Shock (PTS)," SECY-82-465, Nuclear Regulatory Commission, Washington, D. C., November 23, 1982.

A-12. A. L. Lowe, Jr. and J. W. Pegram, "Correlations for Predicting the Effects of Neutron Radiation on Linde 80 Submerged-Arc Welds," BAW-1803, Revision 1, B&W Nuclear Technologies, Inc., Lynchburg, Virginia, May 1991.

A-13. Letter from R. W. Hernan, Nuclear Regulatory Commission, to T. G. Broughton, GPU Nuclear Corporation, Issuance of Amendment No. 176 for TSCR No. 207 (TAC No. M86085), dated August 16, 1993.

A-14. S. E. Yanichko and V. A. Perone, "Analysis of Capsule V from the Virginia Electric Power Company Surry Unit 1 Reactor Vessel Radiation Surveillance Program," WCAP-11415, Westinghouse Electric Corporation, Pittsburgh, Pennsylvania, February 1987.

A-15. S. L. Anderson and A. H. Fero, "Reactor Cavity Neutron Measurement Program for Wisconsin Electric Power Company Point Beach Unit 1," WCAP-12794, Revision 2, Westinghouse Electric Corporation, Pittsburgh, Pennsylvania, December 1992.

A-16. C. A. Ouellette, "Materials Information for Westinghouse-Designed Reactor Vessels Fabricated by B&W," BAW-2150, B&W Nuclear Technologies, Inc., Lynchburg, Virginia, December 1990.

A-17. L. B. Gross, "Chemical Composition of B&W Fabricated Reactor Vessel Beltline Welds," BAW-2121P, B&W Nuclear Technologies, Inc., Lynchburg, Virginia, April 1991.

A-18. K. K. Yoon, "Low Upper-Shelf Toughness Fracture Analysis of Reactor Vessels of B&W Owners Group Reactor Vessel Working Group for Level C & D Service Loads," BAW-2178P, B&W Nuclear Technologies, Inc., Lynchburg, Virginia, February 1993.

A-19. K. K. Yoon, "Low Upper-Shelf Toughness Fracture Analysis of Reactor Vessels of B&W Owners Group Reactor Vessel Working Group for Load Level A & B Conditions," BAW-2192P, Revision 1, B&W Nuclear Technologies Inc., Lynchburg, Virginia, December 1993.

A-20. Code of Federal Regulations, Title 10, Part 50, "Domestic Licensing of Production and Utilization Facilities," Appendix G, Fracture Toughness Requirements.

A-21. S. E. Yanichko and V. A. Perone, "Analysis of Capsule V from the Virginia Electric Power Company Surry Unit 2 Reactor Vessel Radiation Surveillance Program," WCAP-11499, Westinghouse Electric Corporation, Pittsburgh, Pennsylvania, June 1987.