VIRGINIA ELECTRIC AND POWER COMPANY
RICHMOND, VIRGINIA 23261

April 27, 2001

United States Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, D.C. 20555

Serial No. 01-262
NL&OS/ETS
Docket No. 50-339
License No. NPF-7

Gentlemen:

VIRGINIA ELECTRIC AND POWER COMPANY
NORTH ANNA POWER STATION UNIT 2
APPLICATION OF SEQUOYAH 2 SURVEILLANCE DATA TO NORTH ANNA UNIT 2 REACTOR VESSEL WELD MATERIAL FABRICATED FROM WELD WIRE HEAT 4278

Virginia Electric and Power Company (Dominion) provided updates to the NRC's Reactor Vessel Integrity Database (RVID) by letters dated November 19, 1999 and September 19, 2000. The updates considered available reactor vessel materials surveillance data, including data obtained from the North Anna Units 1 and 2 plant-specific surveillance program as well as from other utilities' surveillance programs. During review of proposed changes to the North Anna Units 1 and 2 Technical Specifications for Reactor Coolant System (RCS) pressure/temperature (P/T) operating limits, Low Temperature Overpressure Protection System (LTOPS) setpoints, and LTOPS enabling temperatures (T_{enable}), the NRC reviewer noted that Sequoyah Unit 2 surveillance data had been applied to the North Anna 1 Nozzle-to-Intermediate Shell Weld 05B (ID 6%), but not to the North Anna 2 Nozzle-to-Intermediate Shell Weld 05A (OD 94%). Both of these welds were fabricated with weld wire heat number 4278. The NRC reviewer agreed that the North Anna Units 1 and 2 Nozzle-to-Intermediate Shell Welds were non-limiting materials in terms of their Reference Temperatures for the Nil Ductility Transition (RT_{NDT}), but requested that Dominion provide updated RVID data tables that included explicit consideration of the Sequoyah 2 surveillance data for the North Anna Unit 2 weld fabricated from weld wire heat 4278.

The attachment to this letter provides revised North Anna Unit 2 data tables for the NRC's Reactor Vessel Integrity Database (RVID) and a discussion of changes relative to the previous RVID update for North Anna Unit 2. These evaluations consider the impact of the Sequoyah Unit 2 surveillance data as it applies to the North Anna Unit 2 (a) licensing basis reactor coolant system (RCS) pressure/temperature (P/T) limit curves, (b) the associated Low Temperature Overpressure Protection System (LTOPS) setpoints and enabling temperature, and (c) 10 CFR 50.61 Pressurized Thermal Shock (PTS) screening calculations. As documented in the attached evaluation, the PTS screening calculation results for North Anna Unit 2 continue to meet the applicable screening criteria. Further, the RT_{NDT} value used in the development of the current North Anna Unit 2 Technical Specification P/T limits, LTOPS setpoints, and LTOPS enabling temperature remains conservative.

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By letters dated June 22, 2000, January 4, 2001, and March 22, 2001, a Technical Specification change request was submitted to the NRC for the purpose of modifying the North Anna Units 1 and 2 P/T limits, and extending the cumulative core burnup applicability limits for the existing North Anna Units 1 and 2 LTOPS setpoints and LTOPS enabling temperatures. After consideration of the Sequoyah Unit 2 Capsule W analysis results, it has been determined that the most limiting RT_{NDT} values previously provided to the NRC by letters dated November 19, 1999 and September 19, 2000 remain valid and conservative and continue to support the aforementioned Technical Specification change submittal.

If you have further questions or require additional information, please contact us.

Very truly yours,

Leslie N. Hartz
Vice President - Nuclear Engineering and Services

Commitments made in this letter:

None

Attachment: Evaluation of Application of Sequoyah 2 Surveillance Data to North Anna Unit 2 Reactor Vessel Weld Material Fabricated from Weld Wire Heat 4278

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ATTACHMENT

EVALUATION OF APPLICATION OF SEQUOYAH 2 SURVEILLANCE DATA TO NORTH ANNA UNIT 2 REACTOR VESSEL WELD MATERIAL FABRICATED FROM WELD WIRE HEAT 4278

EVALUATION OF APPLICATION OF SEQUOYAH 2 SURVEILLANCE DATA TO NORTH ANNA UNIT 2 REACTOR VESSEL WELD MATERIAL FABRICATED FROM WELD WIRE HEAT 4278

BACKGROUND

Virginia Electric and Power Company (Dominion) provided updates to the NRC's Reactor Vessel Integrity Database (RVID) by letters dated November 19, 1999 (1) and September 19, 2000 (2). The updates considered available reactor vessel materials surveillance data, including data obtained from the North Anna Units 1 and 2 plant-specific surveillance program as well as from other utilities' surveillance programs. During review of proposed changes to the North Anna Units 1 and 2 Technical Specifications for Reactor Coolant System (RCS) pressure/temperature (P/T) operating limits, Low Temperature Overpressure Protection System (LTOPS) setpoints, and LTOPS enabling temperatures (T_{enable}) (3) (4) (5), the NRC reviewer noted that Sequoyah Unit 2 surveillance data had been applied to the North Anna 1 Nozzle-to-Intermediate Shell Weld 05B (ID 6%) (1), but not to the North Anna 2 Nozzle-to-Intermediate Shell Weld 05A (OD 94%) (2). Both of these welds were fabricated with weld wire heat number 4278. The NRC reviewer agreed that the North Anna Units 1 and 2 Nozzle-to-Intermediate Shell Welds were non-limiting materials in terms of their Reference Temperatures for the Nil Ductility Transition (RT_{NDT}), but requested that Dominion provide updated RVID data tables that included explicit consideration of the Sequoyah 2 surveillance data for the North Anna Unit 2 weld fabricated from weld wire heat 4278.

This evaluation provides revised North Anna Unit 2 data tables for the NRC's Reactor Vessel Integrity Database (RVID) and a discussion of changes relative to the previous RVID update for North Anna Unit 2 (2). The evaluation considers the impact of the Sequoyah Unit 2 surveillance data as it applies to the North Anna Unit 2 (a) licensing basis reactor coolant system (RCS) pressure/temperature (P/T) limit curves, (b) the associated Low Temperature Overpressure Protection System (LTOPS) setpoints and enabling temperature, and (c) 10 CFR 50.61 Pressurized Thermal Shock (PTS) screening calculations. The evaluation was performed in a manner consistent with applicable regulatory guidance. Specifically, the calculation of the Reference Temperature for the Nil Ductility Transition (RT_{NDT}) was performed in accordance with Regulatory Guide 1.99 Revision 2 (7), and the regulatory guidance provided in the meeting minutes from the November 12, 1997 NRC/Industry meeting on reactor vessel integrity (8). PTS screening calculations were performed in accordance with 10 CFR 50.61 (6). Evaluation results are presented in a format consistent with the data requirements of the NRC's Reactor Vessel Integrity Database (RVID).

DISCUSSION OF CHANGES TO PREVIOUSLY REPORTED INFORMATION

Revised RVID data tables for North Anna Unit 2 are presented in Appendix A. Shaded cells indicate a changed value relative to those provided in Reference (2). The following changes (relative to Reference (2)) have been incorporated into the revised tables:

North Anna Unit 2 Nozzle-to-Intermediate Shell Weld 05A (OD 94%), Heat No. 4278

- The RG 1.99 Revision 2 Position 2.1 chemistry factor (CF) calculation includes consideration of Sequoyah 2 surveillance program capsule analysis results. The Sequoyah Unit 2 data are documented in Table 7-7 of Reference (9).
- Because the surveillance capsules were irradiated in a single reactor and the surveillance material was derived from a single source, irradiation temperature and chemistry corrections are not applied in the credibility determination. Because the surveillance capsules were not irradiated in the reactor that is being evaluated, and because the beltline and surveillance material chemical compositions are not essentially identical, application of the surveillance results to the corresponding beltline material is subject to irradiation temperature and chemistry corrections. However, the measured surveillance material transition temperature shift values remain unchanged from the values presented in Reference (9) because the North Anna irradiation temperature is higher than that of Sequoyah Unit 2, and because the copper and nickel concentrations of the North Anna Unit 2 beltline material are less than or equal to the copper and nickel concentrations of the Sequoyah Unit 2 surveillance material. This treatment is consistent with the guidance presented in Reference (8).
- The surveillance data were determined to be non-credible. Further, the data were not within 2σ of the RG 1.99 Rev. 2 Position 1.1 curve based on a CF for the average surveillance material chemical composition. Therefore, the greater of the Position 1.1 and Position 2.1 CF values is applied to the beltline material with a full margin term. This is consistent with the guidance presented in Reference (8).

EVALUATION OF EXISTING P/T LIMITS AND LTOPS SETPOINTS

RT_{NDT} calculations have been performed for all North Anna Unit 2 reactor vessel beltline materials at end-of-license neutron fluence values corresponding to 34.3 EFPY (12). The results are presented in Appendix A. After consideration of the aforementioned changes to previously reported information, the most limiting 1/4-T RT_{NDT} value for North Anna Unit 2 is 209.4°F at a fluence corresponding to an end-of-license cumulative core burnup of 34.3 EFPY (12). This value is unchanged from that previously provided in Reference (2).

EVALUATION OF PTS SCREENING CALCULATIONS

PTS screening calculations have been performed for all North Anna Unit 2 reactor vessel beltline materials at end-of-license neutron fluence values corresponding to 34.3 EFPY (12). The results of these calculations are presented in Appendix A. After consideration of

the aforementioned changes to previously reported information, it is concluded that all North Anna Unit 2 beltline materials continue to meet the 10 CFR 50.61 screening criteria.

CONCLUSIONS

After consideration of the aforementioned changes to previously reported information, the most limiting 1/4-T RT_{NDT} value for North Anna Unit 2 is 209.4°F at a fluence corresponding to an end-of-license cumulative core burnup of 34.3 EFPY (12). This value is unchanged from that previously provided in Reference (2). Thus, the conclusions of Reference (2) remain unchanged. Specifically, the existing North Anna Unit 2 RCS P/T limits, LTOPS setpoints, and LTOPS enabling temperature (10) (11) remain valid and conservative. The proposed revised North Anna Unit 2 RCS P/T limits, LTOPS setpoints, and LTOPS enabling temperature documented in References (3), (4), and (5) are valid to 34.3 EFPY. After consideration of the aforementioned changes to previously reported information, the limiting value of RT_{NDT} for the North Anna Unit 2 reactor vessel beltline materials at 34.3 EFPY remains bounded by the RT_{NDT} value assumed in the proposed revised licensing basis analyses (3) (4) (5). Therefore, the analyses supporting the proposed revised North Anna Unit 2 RCS P/T limits, LTOPS setpoints, and LTOPS enabling temperature remain valid and conservative. Finally, after consideration of the aforementioned changes to previously reported information, all North Anna Unit 2 reactor vessel beltline materials continue to meet the 10 CFR 50.61 PTS screening criteria for cumulative core burnups up to 34.3 EFPY (end-of-license).

NRC REACTOR VESSEL INTEGRITY DATABASE

Virginia Power requests that information presented in Appendix A be used to update the NRC Reactor Vessel Integrity Database (RVID).

References

(1) Letter from L. N. Hartz to USNRC, "Virginia Electric and Power Company, North Anna Power Station Units 1 and 2, Surry Power Station Uints 1 and 2, Evaluation of Reactor Vessel Materials Surveillance Data," dated November 19, 1999.

(2) Letter from L. N. Hartz to USNRC, "Virginia Electric and Power Company, North Anna Power Station Unit 2, Evaluation of Reactor Vessel Materials Surveillance Data," dated September 19, 2000.

(3) Letter from D. A. Christian to USNRC, "Virginia Electric and Power Company, North Anna Power Station Units 1 and 2, Proposed Technical Specifications Changes, Requests for Exemption per 10 CFR 50.60(b), Reactor Coolant System Pressure/Temperature Limits, LTOPS Setpoints, and LTOPS Enable Temperatures," Serial No. 00-306, dated June 22, 2000.

(4) Letter from W. R. Matthews to USNRC, "Virginia Electric and Power Company, North Anna Power Station Units 1 and 2, Response to Request for Additional Information, Proposed Technical Specifications Changes, Reactor Coolant System Pressure/Temperature Limits, LTOPS Setpoints, and LTOPS Enable Temperatures," Serial No. 01-020, dated January 4, 2001.

(5) Letter from W. R. Matthews to USNRC, "Virginia Electric and Power Company, North Anna Power Station Units 1 and 2, Response to Request for Additional Information, Proposed Technical Specifications Changes, Reactor Coolant System Pressure/Temperature Limits, LTOPS Setpoints, and LTOPS Enable Temperatures," Serial No. 01-168, dated March 22, 2001.

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

(7) Regulatory Guide 1.99 Revision 2, "Radiation Embrittlement of Reactor Vessel Materials," dated May, 1988.

(8) Memorandum from K. R. Wichman to E. J. Sullivan, "Meeting Summary for November 12, 1997 Meeting with Owners Group Representatives and NEI Regarding Review of Responses to Generic Letter 92-01, Revision 1, Supplement 1 Responses," dated November 19, 1997.

(9) BAW-2356 Revision 1, "Analysis of Capsule W, Virginia Power North Anna Unit No. 1 Nuclear Power Plant, Reactor Vessel Material Surveillance Program," dated November 1999.

References (continued)

(10) Letter from J. P. O'Hanlon (Virginia Power) to USNRC, "Virginia Electric and Power Company, North Anna Power Station Units 1 and 2, Proposed Technical Specifications Change," dated April 15, 1994 (Virginia Power Serial No. 94-238).

(11) Letter from L. B. Engle (USNRC) to J. P. O'Hanlon, "North Anna Units 1 and 2 – Issuance of Amendments Re: Pressure/Temperature Operating Limits/Low Temperature Overpressure Protection System Pressure Setpoints/Limiting Conditions for Operation, Action Statements, and Surveillance Requirements for PORVs and Block Valves to Address Generic Letter 90-06 (TAC Nos. M77363, M77364, M77433, M77434, M89312, and M89313)," dated October 5, 1994 (Virginia Power Serial No. 94-607).

(12) Letter from N. Kalyanam (USNRC) to J. P. O'Hanlon (Virginia Power), "North Anna Power Station, Units 1 and 2, and Surry Power Station, Units 1 and 2 – Reactor Vessel Fluence Analysis Methodology (Generic Letter 92-01, Revision 1, Supplement 1) (TAC Nos. MA0555, MA0556, MA0576, and MA0577)," dated April 13, 1999 (Virginia Power Serial No. 99-242; Safety Evaluation Report for Virginia Power Topical Report VEP-NAF-3, "Reactor Vessel Fluence Analysis Methodology," dated November, 1997).

APPENDIX A

REACTOR VESSEL MATERIALS DATA TABLES FOR NORTH ANNA UNIT 2

(3 pages)

Facility: North Anna Unit 1
Vessel Manufacturer: Rotterdam Dockyard

RPV Weld Wire Heat or Material ID	Location	Best-Estimate Copper (wt%)	Best-Estimate Nickel (wt%)	EOL ID Fluence (x1E19)	Assigned Material Chemistry Factor	Method of Determining CF	Initial RT(NDT)	Sigma(I)	Sigma(delta)	Margin	Inner Surf. ART or RT(PTS) at EOL	1/4-T ART	Current Licensing Basis 1/4-T ART *
990286/295213	Nozzle Shell Forging	0.160	0.740	0.136	121.5	Tables	6	30.0	17.0	69.0	133.4	121.7	140.3
990311/298244	Intermediate Shell Forging	0.120	0.820	3.920	86.0	Tables	17	0.0	17.0	34.0	167.3	157.7	158.1
990400/292332	Lower Shell Forging	0.156	0.817	3.920	82.9	Surv. Data	38	0.0	17.0	34.0	184.1	174.9	146.6
25295	Nozzle to Int. Shell Circ Weld (0D 94%)	0.352	0.125	0.136	144.2	Surv. Data	0	20.0	14.0	48.8	118.2	104.3	143.2
4278	Nozzle to Int. Shell Circ Weld (ID 6%)	0.120	0.110	0.136	92.4	Surv. Data	0	20.0	22.2	59.8	104.2	95.3	86.6
25531	Int. to Lower Shell Circ Weld	0.098	0.124	3.920	56.2	Tables	19	0.0	28.0	56.0	151.0	144.8	162.9 *

* 1/4-T ART value of 162.9 F was used in the determination of P/T limits

Facility: North Anna Unit 2
Vessel Manufacturer: Rotterdam Dockyard

RPV Weld Wire Heat or Material ID	Location	Best-Estimate Copper (wt%)	Best-Estimate Nickel (wt%)	EOL ID Fluence (x1E19)	Assigned Material Chemistry Factor	Method of Determining CF	Initial RT(NDT)	Sigma(I)	Sigma(delta)	Margin	Inner Surf. ART or RT(PTS) at EOL	1/4-T ART	Current Licensing Basis 1/4-T ART *
990598/291396	Nozzle Shell Forging	0.080	0.770	0.148	51.0	Tables	9	30.0	12.7	65.2	99.7	94.6	96.3
990496/292424	Intermediate Shell Forging	0.107	0.857	3.960	54.1	Surv. Data	75	0.0	17.0	34.0	182.2	176.3	173.3
990533/297355	Lower Shell Forging	0.130	0.830	3.960	96.0	Tables	56	0.0	17.0	34.0	220.0	209.4	196.5*
4278	Nozzle to Int. Shell Circ Weld (0D 94%)	0.120	0.110	0.148	92.4	Surv. Data	0	20.0	23.1	61.1	107.2	98.1	97.6
801	Nozzle to Int. Shell Circ Weld (ID 6%)	0.180	0.110	0.148	87.8	Tables	0	20.0	21.9	59.4	103.3	94.6	75.2
716126	Int. to Lower Shell Circ Weld	0.066	0.046	3.960	26.8	Surv. Data	-48	0.0	14.0	28.0	16.3	13.4	61.4

* 1/4-T ART value of 196.5 F was used in the determination of P/T limits

Note: Shaded cells indicate a changed value relative to values previously submitted in Letter from L. N. Hartz to USNRC, "Virginia Electric and Power Company, North Anna Power Station Unit 2, Evaluation of Reactor Vessel Materials Surveillance Data," dated September 19, 2000.

Table 2: **North Anna 2 Nozzle-to-Intermediate Shell Weld 05A (OD 94%), Heat No. 4278**

Capsule ID (Including Source)	Copper (wt%)	Nickel (wt%)	Irradiation Temperature (F)	Fluence (x1E19)	Measured Delta-RT(NDT) (F)	Data Used In Assessing Vessel? (Yes or No)
Sequoyah Unit 2 Capsule T	0.130	0.110	545.0	0.242	81	Yes
Sequoyah Unit 2 Capsule U	0.130	0.110	545.0	0.608	154	Yes
Sequoyah Unit 2 Capsule X	0.130	0.110	545.0	1.030	30	Yes
-	-	-	-	-	-	-
-	-	-	-	-	-	-
-	-	-	-	-	-	-
-	-	-	-	-	-	-
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-	-	-	-	-	-	-
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Table 3: **North Anna 2 Nozzle-to-Intermediate Shell Weld 05A (OD 94%), Heat No. 4278**

Capsule ID (Including Source)	Copper (wt%)	Nickel (wt%)	Irradiation Temperature (F)	Fluence Factor	Measured Delta-RT(NDT) (F)	Adjusted Delta-RT(NDT) (F) *	Adjusted - Predicted Delta-RT(NDT) (F) *
Sequoyah Unit 2 Capsule T	0.130	0.110	545.0	0.6159	81	81	20
Sequoyah Unit 2 Capsule U	0.130	0.110	545.0	0.8606	154	154	68
Sequoyah Unit 2 Capsule X	0.130	0.110	545.0	1.0083	30	30	-70
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-	-	-	-	-	-	-	-
-	-	-	-	-	-	-	-
-	-	-	-	-	-	-	-
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** For credibility check, measured shift values are adjusted to average surveillance material chemistry and irradiation temperature as required. See Table 4.*

Table 4: North Anna 2 Nozzle-to-Intermediate Shell Weld 05A (OD 94%), Heat No. 4278

CF Determination

Beltline Material ID	Beltline Material Copper (wt%)	Beltline Material Nickel (wt%)	Irradiation Temperature (F)	Position 1.1 Chemistry Factor	Position 2.1 Chemistry Factor	Surveillance Data Credible or Non-Credible?	If Surv. Data Non-Credible, Verify Conservatism of Position 1.1 CF *	Chemistry Factor Applied to Beltline Material **
4278	0.120	0.110	550.3	63.0	92.4	Non-Credible	0.0	92.4

** Measured shift values are adjusted to the average surveillance material chemistry and irradiation temperature, and are verified to be within 2 sigma of the trend curve based on RG 1.99 Rev. 2 Position 1.1.*

*** If surveillance data are non-credible but the Pos. 1.1 CF is shown to be conservative, the lower of the Pos. 1.1 and Pos. 2.1 chemistry factors is applied to the beltline material with a full margin term.*

If surveillance data are non-credible and the Pos. 1.1 CF is shown to be non-conservative, the greater of the Pos. 1.1 and Pos. 2.1 chemistry factors is applied to the beltline material with a full margin term.

Credibility Assessment

						Conservatism Check for Pos. 1.1 CF when Surv. Data Non-Credible		
Capsule ID (Including Source)	(1) Temperature Correction Applied for Credibility?	(2) Chemistry Correction Applied for Credibility?	Surveillance Data Credible or Non-Credible?	(3) Temperature Correction Applied to Surv. Data for Application to Beltline Material?	(4) Chemistry Correction Applied to Surv. Data for Application to Beltline Material?	Adjusted Delta-RT(NDT) (F) *	Adjusted - Predicted Delta-RT(NDT) (F) *	Are adjusted surveillance data within 2 sigma of the applied CF trend curve? *
Sequoyah Unit 2 Capsule T	No	No	Credible	Yes	Yes	81	39	Conservative
Sequoyah Unit 2 Capsule U	No	No	Non-Credible	Yes	Yes	154	96	Non-Conservative
Sequoyah Unit 2 Capsule X	No	No	Non-Credible	Yes	Yes	30	-38	Conservative
-	-	-	-	-	-	-	-	-
-	-	-	-	-	-	-	-	-
-	-	-	-	-	-	-	-	-
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(1) For the credibility determination, a temperature correction is not applied to measured values of transition temperature shift if applicable surveillance data were irradiated in a single reactor (i.e., were irradiated at a similar temperature).

(2) For the credibility determination, a chemistry correction is not applied to measured values of transition temperature shift if applicable surveillance data were obtained from a single source (i.e., were machined from the same block of material).

(3) For determination of the beltline material chemistry factor, a temperature correction is not applied to measured values of transition temperature shift if applicable surveillance data were irradiated in the reactor vessel which is being evaluated (i.e., were irradiated at a similar temperature). A temperature correction is applied only in the conservative direction.

(4) For determination of the beltline material chemistry factor, a chemistry correction (i.e., ratio procedure) is not applied to measured values of transition temperature shift if the chemical composition of applicable surveillance data is essentially identical to the best-estimate chemical composition of the beltline material being evaluated.