VIRGINIA ELECTRIC AND POWER COMPANY
RICHMOND, VIRGINIA 23261

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

Serial No. 96-084
NL&OS/GDM R0
Docket Nos. 50-280, 50-281
50-338, 50-339
License Nos. DPR-32, DPR-37
NPF-4, NPF-7

Gentlemen:

VIRGINIA ELECTRIC AND POWER COMPANY
SURRY AND NORTH ANNA POWER STATIONS UNITS 1 AND 2
PRESSURIZED THERMAL SHOCK (PTS) SCREENING CALCULATIONS

In our letter dated June 8, 1995 (Serial No. 95-197), we submitted a request for revised Technical Specifications heatup and cooldown curves and Low Temperature Overpressure Protection System (LTOPS) setpoints that were valid to the end-of-license for Surry Units 1 and 2. During the NRC's review of this submittal, the staff determined that an outstanding issue existed regarding the currently docketed PTS reference temperature (RT_{PTS}) data for limiting reactor vessel beltline materials previously provided in our letter dated December 10, 1991 (Serial No. 91-328). This data had been provided in response to the revision to 10 CFR 50.61 PTS rule. Specifically, we indicated in our letter that:

> *...none of the revised RT_{PTS} values exceed the applicable screening criterion prior to end-of-license with the exception of that of the Surry Unit 1 Lower Shell Longitudinal Weld L2. As indicated in previous correspondence (Letters Serial Nos. 89-748 dated December 1, 1989, 90-335 dated July 30, 1990, and 91-374 dated July 8, 1991), we are in the process of implementing a flux reduction program at Surry 1. As noted in our July 8, 1991 letter, flux suppression inserts (FSIs) are planned to be installed in Surry Unit 1 during Cycle 13. The target fluence in this program is well below that which could cause RT_{PTS} to exceed the screening criterion at EOL. We consider this plan adequate to ensure that the requirements of 10 CFR 50.61 will continue to be met throughout the operating life of the plant.*

The NRC has requested that Virginia Electric and Power Company provide confirmation of Surry's compliance with the requirements of 10 CFR 50.61 throughout the current license period. In response to this request, we have prepared revised PTS screening calculations for Surry Units 1 and 2. These calculations take credit for available plant-specific and B&W Owners Group Master Integrated Reactor Vessel Materials Surveillance Program (MIRVSP) surveillance data, and utilize best-estimate end-of-license neutron fluence values. No credit is taken for the installation of flux suppression inserts in Unit 1 beginning with Cycle 13.

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Because the December 10, 1991 submittal applied to North Anna as well as Surry, revised calculations have also been prepared for North Anna Units 1 and 2. A discussion of the development of the PTS screening calculations for Surry and North Anna is provided in Attachment 1. A summary of surveillance capsule analysis results and revised PTS screening calculations for North Anna and Surry are provided in Attachments 2 and 3, respectively, and constitute our revised licensing basis.

The revised RT_{PTS} values are below the applicable screening criteria, and therefore ensure that the requirements of 10 CFR 50.61 will continue to be met throughout the license periods for both stations.

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

James P. O'Hanlon

James P. O'Hanlon
Senior Vice-President - Nuclear

Attachments

cc: U. S. Nuclear Regulatory Commission
Region 11
101 Marietta Street, N. W.
Suite 2900
Atlanta, Georgia 30323

Mr. M. W. Branch
NRC Senior Resident Inspector
Surry Power Station

Mr. R. P. McWhorter
NRC Senior Resident Inspector
North Anna Power Station

Attachment 1

Discussion of Pressurized Thermal Shock Screening Calculations Surry and North Anna Power Stations Units 1 and 2

1.0 BACKGROUND

Since our December 1991 Pressurized Thermal Shock (PTS) submittal (2), several reports have been prepared and submitted to the NRC which pertain to Surry's compliance with the requirements of 10 CFR 50.61. However, none of these recent submittals constitutes a revised 10 CFR 50.61 licensing basis:

1. BAW-2166 (3) provides the response to Generic Letter (GL) 92-01, Revision 1 (6).

2. BAW-2222 (5) provides the response to an NRC request for additional information (RAI) on Virginia Power's response to GL 92-01.

3. BAW-2257, Revision 1 (7) provides the response to GL 92-01, Supplement 1 (8) for Linde 80 weld materials.

4. BAW-2260 (11) presents the response to Supplement 1 (8) for the Rotterdam weld materials in the Surry Units 1 and 2 reactor vessels.

Virginia Power docketed these reports by letters dated June 29, 1992 (27), June 30, 1994 (9) and November 20, 1995 (10). Because it contained no reactor vessel beltline fluence estimates, the information in BAW-2166 (3) was insufficient to demonstrate compliance with the requirements of 10 CFR 50.61. Although BAW-2222 (5) and BAW-2260 (11) included all requisite inputs for demonstration of compliance with 10 CFR 50.61, revised RT_{PTS} values were not documented in these reports. Revised RT_{PTS} values were determined in BAW-2257, Revision 1 (7) for Linde 80 weld materials. However, BAW-2257, Revision 1 declared that the Regulatory Guide (RG) 1.99, Revision 2, Position 2.1 "ratio procedure" utilized in the report is unnecessary for the Linde 80 class of welds, and that previously submitted reactor vessel integrity evaluations (i.e., Reference (2)) remain valid. Therefore, the values presented in BAW-2257, Revision 1 do not constitute a revised 10 CFR 50.61 licensing basis for Surry Units 1 and 2.

Unlike the Surry responses to GL 92-01, the North Anna response to GL 92-01, Supplement 1 (11) established a revised 10 CFR 50.61 licensing basis for both North Anna Units. Specifically, Reference (10) concluded that the RT_{PTS} value for the limiting North Anna Units 1 and 2 beltline material increased from 227.7°F to 238.9°F. However, like the calculations which support the Reference (2) PTS submittal, the calculations which support this RT_{PTS} value (10) did not take credit for available surveillance data, and utilized unnecessarily conservative neutron fluence values.

Attachment 1 presents summaries of the surveillance capsule analysis results used in chemistry factor calculations. Attachment 2 presents the results of revised PTS screening calculations for Surry and North Anna Units 1 and 2. Detailed explanations of the chemistry factor and PTS screening calculations accompany the calculation summaries.

2.0 CONSIDERATION OF AN ALTERNATE MEASURED VALUE OF INITIAL RT_{NDT} FOR LINDE 80 WELDS

A key input to the determination of RT_{PTS} is the initial (or unirradiated) value of RT_{NDT}. The B&W Owners Group (B&WOG) Reactor Vessel Working Group has prepared a topical report (16) which concludes that a maximum (upper-bound) unirradiated RT_{NDT} of -27°F is justified for all Linde 80 weld materials. Although the method used to determine this unirradiated RT_{NDT} value departs from the method currently prescribed by Paragraph NB-2331 of Section III of the ASME Boiler and Pressure Vessel Code (BPVC), the method was licensed for the Linde 80 weld material WF-70. In BAW-2245, Revision 1 (16), the applicability of the unirradiated RT_{NDT} value determined for WF-70 is extended to all Linde 80 weld materials.

The method for determining the unirradiated RT_{NDT} prescribed by ASME Section III, Paragraph NB-2331, is referenced in 10 CFR 50.61. In order for Virginia Power to utilize the alternate method, an exemption to the requirements of 10 CFR 50.61 per 10 CFR 50.12 is required. Because compliance with the requirements of 10 CFR 50.61 may be demonstrated without the margins provided by BAW-2245, Revision 1 (16), Virginia Electric and Power Company will not pursue an exemption at this time.

3.0 CONSIDERATION OF THE RG 1.99, REVISION 2, POSITION 2.1 RATIO PROCEDURE

The fracture toughness of reactor pressure vessel steel is indexed to the nil-ductility transition reference temperature (RT_{NDT}). Regulatory Guide (RG) 1.99, Revision 2 (17) provides guidance for determining RT_{NDT}. Because the mean chemical composition (copper and nickel) of surveillance materials may differ from the mean chemical composition of beltline materials, Paragraph 2.1 of RG 1.99, Revision 2 requires measured values of ΔRT_{NDT} (i.e., surveillance capsule analysis results) to be multiplied by the ratio of the respective chemistry factors for the surveillance and beltline materials. This "ratio procedure" adjusts observed transition temperature shifts in an effort to make surveillance capsule analysis results indicative of actual reactor vessel beltline embrittlement.

As reported in Reference (7), the B&WOG Reactor Vessel Working Group (RVWG) believes that differences between the chemical compositions of Linde 80 surveillance and beltline weld materials are fully accommodated by the Position 2.1 method without application of the ratio procedure specified therein. Specifically, the RVWG contends that the mean chemical composition of a beltline weld material (i.e., weld heat) is represented by the mean chemical composition of surveillance and original-fabrication test samples. Therefore, when measured values of ΔRT_{NDT} for two or more surveillance capsule results are evaluated in accordance with Position 2.1 without application of the ratio procedure, the resulting calculated chemistry factor is an unbiased estimate of the chemistry factor for the beltline material. Similarly, the standard deviation of the copper and nickel concentrations of a beltline weld material are represented by the standard deviation of copper and nickel concentrations of surveillance and original-fabrication test samples. Therefore, the ratio procedure need only be applied when there is "clear evidence that the copper or nickel content of the surveillance weld differs from that of the vessel weld" (17). No such evidence exists for Linde 80 welds.

Similarly, there is no clear evidence that the copper or nickel content of the North Anna and Surry Rotterdam surveillance welds differ from those of the corresponding vessel welds. Only minimal differences in mean chemical composition, attributable to random variation, are observed in these materials. Furthermore, as documented in the Reference (11) response to GL 92-01, Revision 1, Supplement 1, application of the ratio procedure to North Anna beltline welds or Surry Rotterdam welds would result in less limiting calculated values of RT_{PTS}. Therefore, the ratio procedure is not applied to North Anna beltline weld materials, nor to Surry weld materials fabricated by Rotterdam.

4.0 DETERMINATION OF CHEMISTRY FACTORS USING CREDIBLE SURVEILLANCE DATA

When two or more credible surveillance data sets are available, they may be used to determine the chemistry factor of the corresponding beltline materials in accordance with Position 2.1 of Regulatory Guide 1.99, Revision 2, Position 2.1. Attachment 1 presents a list of the surveillance capsule analysis results used in Position 2.1 chemistry factor calculations. Attachment 1 also presents a description of the method used to calculate chemistry factors using surveillance data, and documents the determination of credibility of the constituent surveillance data sets.

5.0 PTS SCREENING CALCULATIONS

The results of PTS screening calculations for Surry Units 1 and 2 are presented in Calculations 1 through 22 of Attachment 2. The results for North Anna Units 1 and 2 are presented in Calculations 23 through 34 of Attachment 2. With the exception of Calculations 7A and 11A, all calculations are presented as proposed 10 CFR 50.61 licensing basis calculations. Calculations 7A and 11A utilize a revised unirradiated RT_{NDT} value based on transition range fracture toughness testing (16), and are presented for information only.

As the Attachment 2 calculations demonstrate, all Surry and North Anna reactor vessel beltline materials meet the PTS screening criteria prescribed by 10 CFR 50.61 throughout the current license period. The limiting Surry Units 1 and 2 material (SA-1526) exhibits 8% margin to the applicable screening criterion without consideration of the revised -27°F initial RT_{NDT} value based on transition region fracture toughness testing (16). If the revised initial RT_{NDT} value is considered, the margin to the applicable screening criterion for this material is increased to 19%. The limiting North Anna material exhibits 18% margin to the applicable screening criterion.

6.0 SUMMARY AND CONCLUSIONS

A revised set of PTS screening calculations has been prepared. Although the currently docketed screening calculations for Surry and North Anna Units 1 and 2 (2) took no credit for available surveillance data, the revised calculations utilize the most recently obtained plant-specific and integrated surveillance program material properties data.

Revised best-estimate chemical compositions were determined for several North Anna Units 1 and 2 beltline materials in the North Anna response to GL 92-01, Revision 1, Supplement 1 documented in BAW-2260 (11). The revised calculations use the best-estimate chemical compositions most recently determined (or re-confirmed) in BAW-2257, Revision 1 (7) and BAW-2260 (11).

The Reference (2) PTS submittal utilized the more conservative "design basis" fluence values from the most recent surveillance capsule analyses for Surry Unit 1 (12), Surry Unit 2 (13), North Anna Unit 1 (14), and North Anna Unit 2 (15). The revised calculations utilize the best-estimate fluence values from References (12), (13), (14), and (15).

A revised -27°F initial RT_{NDT} value based on transition region fracture toughness testing applicable to Linde 80 welds was determined in Reference (16). This value was applied to the most limiting Linde 80 weld materials to determine the reduction in the calculated RT_{NDT}, and in the applied margin term. These calculations are submitted for information only.

BAW-2257, Revision 1 (7) dismisses the need to apply the Regulatory Guide 1.99, Revision 2, Position 2.1 ratio procedure to Linde 80 weld materials. Reference (7) contends that differences between the chemical compositions of Linde 80 surveillance and beltline weld materials are fully accommodated by the Position 2.1 method without application of the ratio procedure specified therein. Moreover, there is no clear evidence that the copper or nickel content of the surveillance weld differs from that of the vessel weld.

Similarly, there is no clear evidence that the copper or nickel content of the North Anna and Surry Rotterdam surveillance welds differ from those of the corresponding vessel welds. Furthermore, application of the ratio procedure to North Anna beltline welds or Surry Rotterdam welds would result in less limiting calculated values of RT_{PTS} (11). Therefore, the ratio procedure is not applied to North Anna and Surry Rotterdam weld materials.

All Surry and North Anna Units 1 and 2 reactor vessel beltline materials meet the 10 CFR 50.61 screening criteria throughout the current license periods.

7.0 REFERENCES

(1) Letter from J. P. O'Hanlon to USNRC, "Virginia Electric and Power Company, Surry Power Station Units 1 and 2, Request for Exemption - ASME Code Case N-514, Proposed Technical Specifications Change, Revised Pressure/Temperature Limits and LTOPS Setpoint," NRC Letter 95-197, dated June 8, 1995.

(2) Letter from W. L. Stewart to USNRC, "Virginia Electric and Power Company, Surry Power Station Units 1 and 2, North Anna Power Station Units 1 and 2, Revision to 10 CFR 50.61 Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events," Serial No. 91-328, dated December 10, 1991.

(3) "B&W Owners Group Response to Generic Letter 92-01," BAW-2166, dated June, 1992.

(4) "North Anna Units 1 and 2 Response to Closure Letter for NRC Generic Letter 92-01, Revision 1," BAW-2224, dated July, 1994.

(5) "Reactor Vessel Working Group Response to Closure Letters to NRC Generic Letter 92-01, Revision 1," BAW-2222, dated June, 1994.

(6) Letter from J. G. Partlow (USNRC) to All Holders of Operating Licenses or Construction Permits for Nuclear Power Plants (Except Yankee Atomic Electric Company), "Reactor Vessel Structural Integrity, 10 CFR 50.54(f) (Generic Letter 92-01, Revision 1)," dated March 6, 1992.

(7) "B&W Owners Group Reactor Vessel Working Group Response to Generic Letter 92-01, Revision 1, Supplement 1," BAW-2257, Revision 1, dated October, 1995.

(8) Letter from R. P. Zimmerman (USNRC) to All Holders of Operating Licenses or Construction Permits for Nuclear Power Reactors, "NRC Generic Letter 92-01, Revision 1, Supplement 1: Reactor Vessel Structural Integrity," dated May 19, 1995 (Virginia Power Serial No. 95-270).

(9) Letter from J. P. O'Hanlon to USNRC, "Virginia Electric and Power Company, Surry Power Station Units 1 and 2, Generic Letter 92-01, Revision 1, Reactor Vessel Structural Integrity, Request for Additional Information," Serial No. 94-342, dated June 30, 1994.

(10) Letter from J. P. O'Hanlon to USNRC, "Virginia Electric and Power Company, Surry Power Station Units 1 and 2, North Anna Power Station Units 1 and 2, Six-Month Response to Generic Letter 92-01, Revision 1, Supplement 1, Reactor Vessel Structural Integrity," Serial No. 95-270A, dated November 20, 1995.

(11) "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 Materials," BAW-2260, dated October, 1995.

(12) "Analysis of Capsule V from the Virginia Electric and Power Company Surry Unit 1 Reactor Vessel Radiation Surveillance Program," WCAP-11415, Revision 0, dated February, 1987.

(13) "Analysis of Capsule V from the Virginia Electric and Power Company Surry Unit 2 Reactor Vessel Radiation Surveillance Program," WCAP-11499, Revision 0, dated June, 1987.

(14) "Analysis of Capsule U from the Virginia Electric and Power Company North Anna Unit 1 Reactor Vessel Radiation Surveillance Program," WCAP-11777, dated February, 1988.

(15) "Analysis of Capsule U from the Virginia Electric and Power Company North Anna Unit 2, Reactor Vessel Radiation Surveillance Program," WCAP-12497, dated January, 1990.

(16) "Initial RT_{NDT} of Linde 80 Welds Based on Fracture Toughness in the Transition Range," BAW-2245, Revision 1, dated October 1995.

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

(18) "Surry Unit 1 Reactor Vessel Radiation Surveillance Program," WCAP-7723.

(19) "Surry Unit 2 Reactor Vessel Radiation Surveillance Program," WCAP-8085.

(20) "Virginia Electric and Power Company North Anna Unit 1 Reactor Vessel Radiation Surveillance Program," WCAP-8771, dated September, 1976.

(21) "Virginia Electric and Power Company North Anna Unit 2 Reactor Vessel Radiation Surveillance Program," WCAP-8772, dated November, 1974.

(22) "Master Integrated Reactor Vessel Surveillance Program," BAW-1543, Revision 4, dated February 1993. See also Supplement 1, dated February 1993.

(23) "Surry Units 1 and 2 Reactor Vessel Fluence and RT(PTS) Evaluations," WCAP-11015, Revision 1, dated April, 1987.

(24) "Surry Units 1 and 2 Reactor Vessel Fluence and RT(PTS) Evaluations for Consideration of Life Extension," WCAP-11017, Revision 1, dated April, 1987.

(25) "North Anna Units 1 and 2 Reactor Vessel Fluence and RT(PTS) Evaluations," WCAP-11016, Revision 3, dated January, 1988.

(26) "North Anna Unit 1 Surveillance Capsule Withdrawal Schedule, dated July 1993, Virginia Power Contract ER-MI2002, Westinghouse G.O. RM30416; Attachment to VRA-93-107".

(27) Letter from W. L. Stewart to USNRC, "Virginia Electric and Power Company, Surry Power Station Units 1 and 2, Response to Generic Letter 92-01, Reactor Vessel Structural Integrity," Serial No. 92-211, dated June 29, 1992.

Attachment 2

Summary of Surveillance Capsule Analysis Results Used in RG 1.99, Revision 2, Position 2.1 Chemistry Factor Calculations Surry and North Anna Units 1 and 2

Surveillance Capsules Used in Position 2 RT(PTS) Calculations
Surry Units 1 and 2

Surveillance Capsule	Material ID	Applicable to (Beltline Material ID)	Capsule Fluence (E19)	Measured Shift (F)	Measured - Mean RTNDT	Reference
Surry 1 / T	Forging C4415-1	Forging C4415-1 (Surry 1)	0.281	50	-8	"Analysis of Capsule V from the Virginia Electric and Power Company Surry Unit 1 Reactor Vessel Radiation Surveillance Program," **WCAP-11415, Revision 0**, dated February, 1987.
Surry 1 / V	Forging C4415-1	Forging C4415-1 (Surry 1)	1.940	110	5	**WCAP-11415, Revision 0** (Complete reference presented above.)
TMI 2 / LG1	WF-25	SA-1526 (Surry 1)	0.968	222	7	"B&W Owners Group Reactor Vessel Working Group Response to Generic Letter 92-01, Revision 1, Supplement 1," **BAW-2257, Revision 1**, dated October, 1995.
TMI 2 / LG1	SA-1526	SA-1526 (Surry 1)	0.830	182	-24	**BAW-2257, Revision 1** (Complete reference presented above)
CR 3 / LG1	WF-25	SA-1526 (Surry 1)	0.779	214	12	**BAW-2257, Revision 1** (Complete reference presented above)
TMI 1 / E	WF-25	SA-1526 (Surry 1)	0.107	124	31	**BAW-2257, Revision 1** (Complete reference presented above)
TMI 1 / C	WF-25	SA-1526 (Surry 1)	0.866	203	-5	**BAW-2257, Revision 1** (Complete reference presented above)
Surry 1 / T	SA-1526	SA-1526 (Surry 1)	0.281	165	23	**WCAP-11415, Revision 0** (Complete reference presented above.)
Surry 1 / V	SA-1526	SA-1526 (Surry 1)	1.940	240	-16	**WCAP-11415, Revision 0** (Complete reference presented above.)
CR3-LG1	SA-1585	SA-1585 (Surry 1 & 2)	0.510	148	26	**BAW-2257, Revision 1** (Complete reference presented above)
CR3-LG2	SA-1585	SA-1585 (Surry 1 & 2)	1.670	168	-3	**BAW-2257, Revision 1** (Complete reference presented above)
PB1-V	SA-1263	SA-1585 (Surry 1 & 2)	0.502	110	-11	**BAW-2257, Revision 1** (Complete reference presented above)
PB1-S	SA-1263	SA-1585 (Surry 1 & 2)	0.829	165	23	**BAW-2257, Revision 1** (Complete reference presented above)
PB1-R	SA-1263	SA-1585 (Surry 1 & 2)	2.380	165	-20	**BAW-2257, Revision 1** (Complete reference presented above)
PB1-T	SA-1263	SA-1585 (Surry 1 & 2)	2.420	180	-5	**BAW-2257, Revision 1** (Complete reference presented above)
Surry 2 / X	Forging C4339-1	Forging C4339-1 (Surry 2)	0.302	55	9	"Analysis of Capsule V from the Virginia Electric and Power Company Surry Unit 2 Reactor Vessel Radiation Surveillance Program," **WCAP-11499, Revision 0**, dated June, 1987.
Surry 2 / V	Forging C4339-1	Forging C4339-1 (Surry 2)	1.880	75	-5	**WCAP-11499, Revision 0** (Complete reference presented above.)
Surry 2 / X	R3008	R3008 (Surry 2)	0.302	95	9	**WCAP-11499, Revision 0** (Complete reference presented above.)
Surry 2 / V	R3008	R3008 (Surry 2)	1.880	145	-5	**WCAP-11499, Revision 0** (Complete reference presented above.)

Surveillance Capsules Used in Position 2 RT(PTS) Calculations
North Anna Units 1 and 2

Surveillance Capsule	Material ID	Applicable to (Beltline Material ID)	Capsule Fluence (E19)	Measured Shift (F)	Measured - Mean RTNDT	Reference
North Anna 1 / V	Forging 03 (Longitudinal)	Forging 03 (North Anna 1)	0.249	39	-16	"Analysis of Capsule V from the Virginia Electric and Power Company North Anna Unit 1 Reactor Vessel Radiation Surveillance Program," **BAW-1638, Revision 0**, dated May, 1981.
North Anna 1 / U	Forging 03 (Longitudinal)	Forging 03 (North Anna 1)	0.828	95	11	"Analysis of Capsule U from the Virginia Electric and Power Company North Anna Unit 1 Reactor Vessel Radiation Surveillance Program," **WCAP-11777, Revision 0**, dated February, 1988.
North Anna 1 / V	Forging 03 (Transverse)	Forging 03 (North Anna 1)	0.249	21	-15	**BAW-1638, Revision 0** (Complete reference presented above.)
North Anna 1 / U	Forging 03 (Transverse)	Forging 03 (North Anna 1)	0.828	65	10	**WCAP-11777, Revision 0** (Complete reference presented above.)
North Anna 1 / V	Weld 04	Weld 04 (North Anna Unit 1)	0.249	78	20	**BAW-1638, Revision 0** (Complete reference presented above.)
North Anna 1 / U	Weld 04	Weld 04 (North Anna Unit 1)	0.828	75	-13	**WCAP-11777, Revision 0** (Complete reference presented above.)
North Anna 2 / V	Forging 04 (Longitudinal)	Forging 04 (North Anna 2)	0.241	9	-5	"Analysis of Capsule V from the Virginia Electric and Power Company North Anna Unit 2 Reactor Vessel Radiation Surveillance Program," **BAW-1794, Revision 0**, dated October, 1983.
North Anna 2 / U	Forging 04 (Longitudinal)	Forging 04 (North Anna 2)	0.955	25	3	"Analysis of Capsule U from the Virginia Electric and Power Company North Anna Unit 2 Reactor Vessel Radiation Surveillance Program," **WCAP-12497, Revision 0**, dated January, 1990.
North Anna 2 / V	Forging 04 (Transverse)	Forging 04 (North Anna 2)	0.241	9	-20	**BAW-1794, Revision 0** (Complete reference presented above.)
North Anna 2 / U	Forging 04 (Transverse)	Forging 04 (North Anna 2)	0.955	60	13	**WCAP-12497, Revision 0** (Complete reference presented above.)
North Anna 2 / V	Weld 04	Weld 04 (North Anna Unit 2)	0.241	2	-4	**BAW-1794, Revision 0** (Complete reference presented above.)
North Anna 2 / U	Weld 04	Weld 04 (North Anna Unit 2)	0.955	13	3	**WCAP-12497, Revision 0** (Complete reference presented above.)

Surveillance Capsule Identification

The station, unit, and surveillance capsule identifier are presented in the **Surveillance Capsule** column. Additional information on currently docketed plant-specific and integrated surveillance program capsule withdrawal schedules is available in References (26), (15), and (22).

Surveillance Capsule Analysis Results

The surveillance material identifier and the beltline material to which the surveillance capsule material is applicable are presented in the **Material ID** and **Beltline Material ID** columns, respectively. The surveillance capsule fluence (in units of $n/cm^2 \times 10^{19}$) and measured transition temperature shift are presented in the **Capsule Fluence** and **Measured Shift** columns. The **Measured Shift** (ΔRT_{NDT}) is calculated as the difference between the temperatures at which irradiated and unirradiated Charpy specimens exhibit 30 ft-lb of absorbed energy.

Chemistry Factor Calculations

According to Regulatory Guide 1.99, Revision 2, the fluence factor (FF) is defined as:

$$FF = f^{(0.28-0.10 \log f)}$$

where f is the neutron fluence (E>1.0 MeV) accumulated by the surveillance capsule in units of 10^{19} n/cm^2. When surveillance data is available, the chemistry factor is calculated by dividing the sum of all $\Delta RT_{NDT} \times FF$ values by the sum of all FF^2 values. The calculated CF value provides the "best fit" of available surveillance data to an equation of the form:

$$\Delta RT_{NDT} = (CF) f^{(0.28-0.10 \log f)}$$

where CF is the chemistry factor, and f is the neutron fluence in units of 10^{19} n/cm^2, E>1.0 MeV.

Assessment of Surveillance Data Credibility

In consideration of the credibility of available surveillance data, RG 1.99, Revision 2, Section B (Discussion) states:

> *When there are two or more sets of surveillance data from one reactor, the scatter of ΔRT_{NDT} values about a best-fit line drawn as described in Regulatory Guide Position 2.1 normally should be less than 28°F for welds and 17°F for base metal. Even if the fluence range is large (two or more orders of magnitude), the scatter should not exceed twice those values. Even if the data fail this criterion for use in shift calculations, they may be credible for*

determining decrease in upper-shelf energy if the upper shelf can be clearly determined, following the definition given in ASTM E 185-82 (Ref. 1).

The credibility criteria established in RG 1.99, Revision 2 are 1σ values. (RG 1.99, Revision 2, Section 1.1 states *"The standard deviation for ΔRT_{NDT},σ_Δ, is 28°F for welds and 17°F for base metal..."*) Approximately 67% of the data points can be statistically expected to lie within $\pm 1\sigma$, and approximately 95% within $\pm 2\sigma$. All measured ΔRT_{NDT} values used in Position 2.1 chemistry factor calculations are well within $\pm 2\sigma$ scatter bands (i.e., 2 x 28°F for welds; 2 x 17°F for base metal) around the best-fit ΔRT_{NDT} line. Of the 31 surveillance data points considered, only two surveillance data points have ΔRT_{NDT} values which lie beyond $\pm 1\sigma$ scatter bands (i.e., 28°F for welds; 17°F for base metal) around a best-fit ΔRT_{NDT} line. These two data points exceed the $\pm 1\sigma$ credibility criteria by only 3°F. On this basis, all data points used in Position 2 chemistry factor calculations were determined to be credible.

Attachment 3

Revised Pressurized Thermal Shock Screening Calculations
Surry and North Anna Units 1 and 2

Surry Unit 1 PTS Screening Calculations

Calculation Number	Material Identification	Heat Number	Weld ID	wt% Cu	wt% Ni	CF (Pos. 1)	Plant-Spec. Surv. Mtl.?	MIRVSP Surv. Mtl.?	CF Used in Calcs	Inner Surf. Fluence (E19)	Axial Multiplier
1	Nozzle Shell Forging	122V109VA1	n/a	0.09	0.74	58.0	-	-	58.0	3.96	0.12
2	Intermediate Shell Plate	C4326-1	n/a	0.11	0.55	73.5	yes	-	73.5	3.96	1.00
3	Intermediate Shell Plate	C4326-2	n/a	0.11	0.55	73.5	-	-	73.5	3.96	1.00
4	Lower Shell Plate	C4415-1	n/a	0.11	0.50	73.0	yes	-	89.2	3.96	1.00
5	Lower Shell Plate	C4415-2	n/a	0.11	0.50	73.0	-	-	73.0	3.96	1.00
6	Nozzle to Int Shell Circ Weld	25017	J726	0.33	0.10	152.0	-	-	152.0	3.96	0.12
7	Int. to Low Sh. Circ Weld (ID 40%)	72445	SA-1585	0.21	0.59	162.5	-	yes	149.8	3.96	1.00
7A	*Int. to Low Sh. Circ Weld (ID 40%)*	*72445*	*SA-1585*	*0.21*	*0.59*	*162.5*	-	*yes*	*149.8*	*3.96*	*1.00*
8	Int. to Low Sh. Circ Weld (OD 60%)	72445	SA-1650	0.21	0.59	162.5	-	yes	149.8	3.96	1.00
9	Int Shell Long. Welds L3 & L4	8T1554	SA-1494	0.18	0.63	159.0	-	-	159.0	0.639	1.00
10	Lower Shell Long. Weld L1	8T1554	SA-1494	0.18	0.63	159.0	-	-	159.0	0.639	1.00
11	Lower Shell Long. Weld L2	299L44	SA-1526	0.35	0.68	223.6	yes	yes	217.0	0.639	1.00
11A	*Lower Shell Long. Weld L2*	*299L44*	*SA-1526*	*0.35*	*0.68*	*223.6*	*yes*	*yes*	*217.0*	*0.639*	*1.00*

Calculation Number	Adjusted Fluence (E19)	Initial RTNDT	"Measured" or "Estimate"	10 CFR 50.61 Margin Term	Delta RTNDT	RTPTS	Screening Criterion	Ratio (RT_{PTS} /Screen Crit.)
1	0.475	40	Measured	34	46.0	120.0	270	44%
2	3.960	10	Measured	34	99.5	143.5	270	53%
3	3.960	0	Measured	34	99.5	133.5	270	49%
4	3.960	20	Measured	34	120.8	174.8	270	65%
5	3.960	0	Measured	34	98.8	132.8	270	49%
6	0.475	0	Estimate	66	120.5	186.5	300	62%
7	3.960	-5	Estimate	66	202.8	263.8	300	88%
7A	*3.960*	*-27*	*Measured*	*56*	*202.8*	*231.8*	*300*	*77%*
8	3.960	-5	Estimate	66	202.8	263.8	300	88%
9	0.639	-5	Estimate	66	139.0	200.0	270	74%
10	0.639	-5	Estimate	66	139.0	200.0	270	74%
11	0.639	-7	Estimate	66	189.8	248.8	270	92%
11A	*0.639*	*-27*	*Measured*	*56*	*189.8*	*218.8*	*270*	*81%*

Surry Unit 2 PTS Screening Calculations

Calculation Number	Material Identification	Heat Number	Weld ID	wt% Cu	wt% Ni	CF (Pos. 1)	Plant-Spec. Surv. Mtl.?	MIRVSP Surv. Mtl.?	CF Used in Calcs	Inner Surf. Fluence (E19)	Axial Multiplier
12	Nozzle Shell Forging	123V303VA1	n/a	0.09	0.73	58.0	-	-	58.0	3.43	0.12
13	Intermediate Shell Plate	C4331-2	n/a	0.12	0.60	83.0	-	-	83.0	3.43	1.00
14	Intermediate Shell Plate	C4339-2	n/a	0.11	0.54	73.4	-	-	73.4	3.43	1.00
15	Lower Shell Plate	C4208-2	n/a	0.15	0.55	107.3	-	-	107.3	3.43	1.00
16	Lower Shell Plate	C4339-1	n/a	0.11	0.54	73.4	yes	-	68.4	3.43	1.00
17	Nozzle to Int Shell Circ Weld	4275	L737	0.35	0.10	160.5	-	-	160.5	3.43	0.12
18	Int. to Lower Shell Circ Weld	0227	R3008	0.19	0.55	149.3	yes	-	128.0	3.43	1.00
19	Int. Shell Long. Weld L4 (ID 50%)	8T1762	WF-4	0.20	0.55	152.3	-	-	152.3	0.714	1.00
20	Int. Sh. Welds L3 (100%), L4 (OD 50)	72445	SA-1585	0.21	0.59	162.5	-	yes	149.8	0.714	1.00
21	LS Welds L2 (ID 63%), L1 (100)	8T1762	WF-4	0.20	0.55	152.3	-	-	152.3	0.714	1.00
22	LS Long. Weld L2 (OD 37%)	8T1762	WF-8	0.20	0.55	152.3	-	-	152.3	0.714	1.00

Calculation Number	Adjusted Fluence (E19)	Initial RTNDT	"Measured" or "Estimate"	10 CFR 50.61 Margin Term	Delta RTNDT	RTPTS	Screening Criterion	Ratio (RT_{PTS} /Screen Crit.)
12	0.412	30	Measured	34	43.7	107.7	270	40%
13	3.430	-10	Measured	34	109.7	133.7	270	50%
14	3.430	-20	Measured	34	97.0	111.0	270	41%
15	3.430	-30	Measured	34	141.8	145.8	270	54%
16	3.430	-10	Measured	34	90.4	114.4	270	42%
17	0.412	0	Estimate	66	121.0	187.0	300	62%
18	3.430	0	Estimate	66	169.2	235.2	300	78%
19	0.714	-5	Estimate	66	137.9	198.9	270	74%
20	0.714	-5	Estimate	66	135.6	196.6	270	73%
21	0.714	-5	Estimate	66	137.9	198.9	270	74%
22	0.714	-5	Estimate	66	137.9	198.9	270	74%

North Anna Unit 1 PTS Screening Calculations

Calculation Number	Material Identification	Heat Number	Weld ID	wt% Cu	wt% Ni	CF (Pos. 1)	Plant-Spec. Surv. Mtl.?	MIRVSP Surv. Mtl.?	CF Used in Calcs	Inner Surf. Fluence (E19)	Axial Multiplier
23	Nozzle Shell Forging	990286/295213	Forging 05	0.16	0.74	121.5	-	-	121.5	3.95	0.07
24	Intermediate Shell Forging	990311/298244	Forging 04	0.12	0.82	86.0	-	-	86.0	3.95	1.00
25	Lower Shell Forging	990400/292332	Forging 03	0.16	0.83	123.3	yes	-	88.9	3.95	1.00
26	Nozzle to Int. Shell Circ Weld (0D 94%)	25295	Weld 05A	0.35	0.13	162.8	-	-	162.8	3.95	0.07
27	Nozzle to Int. Shell Circ Weld (ID 6%)	4278	Weld 05B	0.12	0.11	63.0	-	-	63.0	3.95	0.07
28	Int. to Lower Shell Circ Weld	25531	Weld 04	0.11	0.13	61.4	yes	-	93.1	3.95	1.00

Calculation Number	Adjusted Fluence (E19)	Initial RTNDT	"Measured" or "Estimate"	10 CFR 50.61 Margin Term	Delta RTNDT	RTPTS	Screening Criterion	Ratio (RT_{PTS} /Screen Crit.)
23	0.277	6	Estimate	48	78.9	132.9	270	49%
24	3.950	17	Measured	34	116.4	167.4	270	62%
25	3.950	38	Measured	34	120.3	192.3	270	71%
26	0.277	0	Estimate	66	105.7	171.7	300	57%
27	0.277	0	Estimate	66	40.9	106.9	300	36%
28	3.950	19	Measured	56	126.0	201.0	300	67%

North Anna Unit 2 PTS Screening Calculations

Calculation Number	Material Identification	Heat Number	Weld ID	wt% Cu	wt% Ni	CF (Pos. 1)	Plant-Spec. Surv. Mtl.?	MIRVSP Surv. Mtl.?	CF Used in Calcs	Inner Surf. Fluence (E19)	Axial Multiplier
29	Nozzle Shell Forging	990598/291396	Forging 05	0.08	0.77	51.0	-	-	51.0	4.47	0.07
30	Intermediate Shell Forging	990496/292424	Forging 04	0.10	0.85	67.0	yes	-	47.9	4.47	1.00
31	Lower Shell Forging	990533/297355	Forging 03	0.13	0.83	96.0	-	-	96.0	4.47	1.00
32	Nozzle to Int. Shell Circ Weld (0D 94%)	4278	Weld 05A	0.12	0.11	63.0	-	-	63.0	4.47	0.07
33	Nozzle to Int. Shell Circ Weld (ID 6%)	801	Weld 05B	0.18	0.11	87.8	-	-	87.8	4.47	0.07
34	Int. to Lower Shell Circ Weld	716126	Weld 04	0.07	0.05	37.8	yes	-	10.4	4.47	1.00

Calculation Number	Adjusted Fluence (E19)	Initial RTNDT	"Measured" or "Estimate"	10 CFR 50.61 Margin Term	Delta RTNDT	RTPTS	Screening Criterion	Ratio (RT_{PTS} /Screen Crit.)
29	Forging	Forging 05	990598/291396	48	34.7	91.7	270	34%
30	Forging	Forging 04	990496/292424	34	66.1	175.1	270	65%
31	Forging	Forging 03	990533/297355	34	132.5	222.5	270	82%
32	Weld	Weld 05A	4278	66	42.9	108.9	300	36%
33	Weld	Weld 05B	801	66	59.8	125.8	300	42%
34	Weld	Weld 04	716126	56	14.3	22.3	300	7%

Material Identification

Each reactor vessel beltline material is identified by a **Material Identification, Heat Number**, and **Weld ID** (if applicable). The information which identifies each beltline material is consistent with the descriptions in the Surry and North Anna responses to GL 92-01, Revision 1 closure letters (4),(5), and the Surry and North Anna responses to GL 92-01, Revision 1, Supplement 1 (7),(11).

Chemical Composition

The beltline material mean chemical compositions **(wt% Cu and wt% Ni)** for the Surry units are documented in the responses to the NRC requests for additional information on the responses to GL 92-01, Revision 1 (5). The Linde 80 weld mean chemical compositions were re-verified in the B&W Owners Group response to GL 92-01, Revision 1, Supplement 1 (7). The mean chemical compositions for all North Anna Units 1 and 2 beltline materials, and the Surry Units 1 and 2 Rotterdam weld materials were revised (or re-verified) in the Reference (11) response to GL 92-01, Revision 1, Supplement 1. The 10 CFR 50.61 and Position 1 of RG 1.99, Revision 2 beltline material chemistry factors **(CF (Pos. 1))** were determined using these mean chemical compositions.

Chemistry Factors Used in Screening Calculations

If a material is included in a plant-specific surveillance program (18),(19),(20),(21) or in the B&W Owners Group Master Integrated Reactor Vessel Surveillance Program (MIRVSP) (22), this is indicated in the **Plant-Specific Surveillance Material** or **MIRVSP Surveillance Material** columns. If two or more credible surveillance capsule analysis results are available for a particular beltline material, the chemistry factor calculated in accordance with Regulatory Guide 1.99, Revision 2, Paragraph 2.1 ("Position 2") is used in the PTS screening calculation. (Compare **CF Used in Calculations** to **CF (Pos. 1)**.) The credibility assessment of surveillance capsule analysis results is described in Attachment 1.

Neutron Fluence

The maximum **Inner Surface Neutron Fluence** values (in units of $n/cm^2 \times 10^{19}$) were obtained from References (12), (13), (14), and (15). The Surry Unit 1 fluence estimates do not take credit for the installation of flux suppression inserts.

Correction for Axial Location of Beltline Material

The **Axial Multiplier** corrects the maximum **Inner Surface Neutron Fluence** for axial elevations other than the core mid-plane. The product of the **Axial Multiplier** and the maximum **Inner Surface Neutron Fluence** is presented in the **Adjusted Fluence** column.

The **Axial Multiplier** for the Surry units is obtained from Figure II.2-6 of Reference (23) or Figure 6-7 of References (12) and (13). The **Axial Multiplier** for the North Anna units is obtained from Figure II.2-6 of Reference (25) or Figure 6-7 of Reference (14). The Surry Units 1 and 2 nozzle shell forging to intermediate shell circumferential weld is located 9.0 inches above the top of the active region of the core, or 81.0 inches (206.0 cm) above the core midplane. (See Figures III.3-1 and III.3-2 of Reference (24).) For the North Anna Units, the nozzle shell forging to intermediate shell circumferential weld is located 13.6 inches above the top of the active region of the core, or 85.6 inches (217.0 cm) above the core midplane. (See Figures III.1-1 and III.1-2 of Reference (25).) These values (206.0 cm and 217.0 cm) also represent conservative estimates of the axial location of the nozzle shell forging.

Unirradiated RT_{NDT}

The **Initial RT_{NDT}** column presents the unirradiated nil-ductility transition reference temperature, calculated in accordance with the requirements of the ASME Code. Values were obtained from BAW-2222 (5) and BAW-2260 (11). Because the **10 CFR 50.61 Margin Term** is dependent on the method of determining the **Initial RT_{NDT}**, each **Initial RT_{NDT}** value is flagged as being either a **Measured or Estimated** value. Supplemental calculations were performed for the limiting Surry Units 1 and 2 Linde 80 weld materials (i.e., Unit 1 Intermediate to Lower Shell Circumferential Weld, ID-40%, SA-1585; and Unit 1 Lower Shell Longitudinal Weld L2, SA-1526) using the measured -27°F **Initial RT_{NDT}** value determined in Reference (16). As previously noted, calculations using the Reference (16) **Initial RT_{NDT}** value are presented for information and comment only.

10 CFR 50.61 Margin Term

The 10 CFR 50.61 PTS Rule prescribes **Margin Term** values for uncertainty in the **Initial RT_{NDT}** and in the **Delta RT_{NDT}** due to irradiation. For welds, the **Margin Term** values are 56°F and 66°F for **Measured** and **Estimated** values of **Initial RT_{NDT}**, respectively. For plates and forgings, the **Margin Term** values are 34°F and 48°F for **Measured** and **Estimated** values of **Initial RT_{NDT}**, respectively.

RT_{NDT} Shift Due to Irradiation (Delta RT_{NDT})

Delta RT_{NDT} is determined in accordance with the methodology prescribed in 10 CFR 50.61:

$$\Delta RT_{NDT} = (CF)f^{(0.28 - 0.10\log f)}$$

where **CF** is the **Chemistry Factor** described above, and f is the **Neutron Fluence** in units of 10^{19} n/cm^2. **RT_{PTS}** is the sum of the **Initial RT_{NDT}**, the **Margin Term**, and **Delta RT_{NDT}**. **RT_{PTS}** values are compared to the applicable **Screening Criterion** specified by 10 CFR 50.61. For circumferential welds, the **Screening Criterion** is 300°F. For axial (longitudinal) welds, plates, and forgings, the **Screening Criterion** value is 270°F. The **Ratio of RT_{PTS} to the Screening Criterion** permits ranking of beltline materials in terms of the relative proximity of a specific material's **RT_{PTS}** to the applicable **Screening Criterion**. The most limiting materials have the highest ratio.