VIRGINIA ELECTRIC AND POWER COMPANY
RICHMOND, VIRGINIA 23261

September 28, 1992

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

Serial No. 92-211B
NL&P/CGL R2
Docket Nos. 50-338
50-339
License Nos. NFP-4
NFP-37

Gentlemen:

**VIRGINIA ELECTRIC AND POWER COMPANY**
**NORTH ANNA POWER STATION UNITS 1 AND 2**
**SUPPLEMENT TO RESPONSE TO GENERIC LETTER 92-01**
**REACTOR VESSEL STRUCTURAL INTEGRITY**

In a June 29, 1992 letter (Serial No. 92-211A), Virginia Electric and Power Company provided the response to Generic Letter 92-01 for North Anna Units 1 and 2. This information was documented in B&W Nuclear Service Company Report BAW-2168, dated June 1992 and entitled "Response to Generic Letter 92-01 for Virginia Electric and Power Company North Anna Unit 1 and North Anna Unit 2."

During a September 18, 1992 conference call, Mr. Barry Elliott of the NRC requested a copy of B&W Report BAW-1911 Revision 1, which is referenced in Report BAW-2168. In reply to that request, enclosed is a copy of B&W Report BAW-1911 Revision 1, dated August 1986 and entitled "Reactor Pressure Vessel and Surveillance Program Materials Licensing Information for North Anna Units 1 and 2."

Also related to our June 29, 1992 transmittal, the following information is provided. During a follow-up review of the supporting calculations for Report BAW-2168, B&W determined that a discrepancy existed in the Units 1 and 2 Table 10 Columns 3 and 6. This discrepancy consisted of B&W's use of the reported copper and nickel chemistry contents for the beltline materials for Forging 03 (Unit 1) and Forging 04 (Unit 2) in the calculation of the predicted $\Delta RT_{NDT} + 2\sigma$ and the predicted $C_V USE$ in lieu of the contents for the surveillance materials. Since the chemical composition differences between the beltline forgings and the surveillance materials are small, the resulting changes are insignificant. Specifically, the resulting changes are:

- In the North Anna Unit 1 Table 10 for Forging 03, the Column 3 values of Predicted $\Delta RT_{NDT} + 2\sigma$ changed from 106°F and 143°F to 110°F and 150°F, respectively.

- In the North Anna Unit 2 Table 10 for Forging 04, the Column 3 values of Predicted $\Delta RT_{NDT} + 2\sigma$ changed from 70°F and 91°F to 81°F and 110°F, respectively.

- In the North Anna Unit 2 Table 10 for Forging 04, the Column 6 values of Predicted $C_V$USE Drop changed from 9 ft-lb and 12 ft-lb to 10 ft-lb and 14 ft-lb, respectively.

Revised pages reflecting these changes are attached for temporary insertion into the North Anna report. A future transmittal will provide a Revision 1 version of Report BAW-2168 including these changes. The conclusion reached in our response to Generic Letter 92-01 for North Anna remains valid and is that North Anna continues to be in compliance with applicable regulations governing reactor vessel integrity.

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

Very truly yours.

W. L. Stewart
Senior Vice President - Nuclear

Enclosure - B&W Report BAW-1911 Revision 1, "Reactor Pressure Vessel and Surveillance Program Materials Licensing Information for North Anna Units 1 and 2

Attachments: Revised Table 10 from BAW-2168 for North Anna Unit 1
Revised Table 10 from BAW-2168 for North Anna Unit 2

cc: U. S. Nuclear Regulatory Commission
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COMMONWEALTH OF VIRGINIA )
)
COUNTY OF HENRICO )

The foregoing document was acknowledged before me, in and for the County and Commonwealth aforesaid, today by E. W. Harrell, who is Vice President - Nuclear Engineering Services, for W. L. Stewart who is Senior Vice President - Nuclear, of Virginia Electric and Power Company. He is duly authorized to execute and file the foregoing document in behalf of that Company, and the statements in the document are true to the best of his knowledge and belief.

Acknowledged before me this 28 day of September, 1992.

My Commission Expires: May 31, 1994.

Vicki L. Hull
Notary Public

(SEAL)

TABLE 10. GENERIC LETTER 92-01 RESPONSE: SECTION 3, ITEM c

Subject: Generic Letter 88-11 Response Commitments; Difference Between Measured and Predicted (Regulatory Guide 1.99, Revision 2) Embrittlement Effects

Plant: North Anna Unit 1

Question I. Does measured $\Delta RT_{NDT}$ exceed $\Delta RT_{NDT} + 2\sigma$ predicted by Regulatory Guide 1.99, Revision 2?

Question II. Does measured $C_v$USE drop exceed that obtained from Regulatory Guide 1.99, Revision 2, Figure 2?

| Column 1 | Column 2 | Column 3 | Column 4 | Column 5 | Column 6 | Column 7 |
|---|---|---|---|---|---|---|
| Beltline Material | Measured $\Delta RT_{NDT}$ | Predicted $\Delta RT_{NDT} + 2\sigma$ | Question I Yes/No | Measured $C_v$USE Drop | Predicted $C_v$USE Drop | Question II Yes/No |
| Forging 05 | ND | ND | NA | ND | ND | NA |
| Forging 04 | ND | ND | NA | ND | ND | NA |
| Forging 03 | 39(1) | 110 | No | 25(1) | 25 | Yes(4) |
| | 65(2) | 150 | No | 0(2) | 20 | No |
| Weld 05A | ND | ND | NA | ND | ND | NA |
| Weld 05B | ND | ND | NA | ND | ND | NA |
| Weld 04 | 78(1) | 64 | Yes(3) | 3(1) | 15 | No |
| | 75(2) | 98 | No | 3(2) | 21 | No |

NOTES FOR TABLE 10 ARE ON THE FOLLOWING PAGE.

TABLE 10 (CONTINUED)

NOTES TO TABLE 10:

(1) BAW-1638

(2) WCAP-11777

(3) The only instance where a measured "shift" exceeds that predicted by calculation performed in accordance with Regulatory Guide 1.99, Revision 2, is for weld metal. This result was obtained for surveillance material representing Weld 04 irradiated to $2.49 \times 10^{18}$ nvt. This result was not observed for surveillance material irradiated to $8.28 \times 10^{18}$ nvt. Since the measured shift of the material irradiated to a higher neutron fluence did not exceed the predicted value, it is safe to conclude that the conservativeness of the Regulatory Guide method was not compromised.

(4) The only instance the measured "drop" in C$_v$USE exceeded that predicted by calculation performed in accordance with Regulatory Guide 1.99, Revision 2 is for Forging 03. This result was not found for the same material (Forging 03) at a higher fluence and the requirements of 10CFR50, Appendix G, were not violated. Because the "drop" data did not violate regulatory requirements, and because there is no further application of the "drop" data, it is concluded that the effect of these surveillance results is not significant.

TABLE 10. GENERIC LETTER 92-01 RESPONSE: SECTION 3, ITEM c

Subject: Generic Letter 88-11 Response Commitments; Difference Between Measured and Predicted (Regulatory Guide 1.99, Revision 2) Embrittlement Effects

Plant: North Anna Unit 2

Question I. Does measured $\Delta RT_{NDT}$ exceed $\Delta RT_{NDT}$ + 2$\sigma$ predicted by Regulatory Guide 1.99, Revision 2?

Question II. Does measured $C_v$USE drop exceed that obtained from Regulatory Guide 1.99, Revision 2, Figure 2?

| Column 1 | Column 2 | Column 3 | Column 4 | Column 5 | Column 6 | Column 7 |
|---|---|---|---|---|---|---|
| Beltline Material | Measured $\Delta RT_{NDT}$ | Predicted $\Delta RT_{NDT}$ + 2$\sigma$ | Question I Yes/No | Measured $C_v$USE Drop | Predicted $C_v$USE Drop | Question II Yes/No |
| Forging 05 | ND | ND | NA | ND | ND | NA |
| Forging 04 | 9(1) | 81 | No | 13(1) | 10 | Yes(3) |
| | 60(2) | 110 | No | 0(2) | 14 | No |
| Forging 03 | ND | ND | NA | ND | ND | NA |
| Weld 05A | ND | ND | NA | ND | ND | NA |
| Weld 05B | ND | ND | NA | ND | ND | NA |
| Weld 04 | 2(1) | 60 | No | 23(1) | 18 | Yes(3) |
| | 13(2) | 96 | No | 0(2) | 25 | No |

NOTES FOR TABLE 10 ARE ON THE FOLLOWING PAGE.

TABLE 10 (CONTINUED)

NOTES TO TABLE 10:

(1) BAW-1794

(2) WCAP-12497

(3) The only instances the measured "drop" in $C_v$USE exceeded that predicted by calculation performed in accordance with Regulatory Guide 1.99, Revision 2, is for Forging 04 and Weld 04. This result was not found for the same material (Forging 04 and Weld 04) at a higher fluence and the requirements of 10CFR50, Appendix G, were not violated. Because the "drop" data did not violate regulatory requirements, and because there is no further application of the "drop" data, it is concluded that the effect of these surveillance results is not significant.

BAW-1911, Rev. 1
August 1986

# REACTOR PRESSURE VESSEL AND SURVEILLANCE PROGRAM MATERIALS LICENSING INFORMATION FOR NORTH ANNA UNITS 1 AND 2

by
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B&W Control No. 77-1163804-01
B&W Contract No. 583-7375, Task 042

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## RECORD OF REVISIONS

| Date | Revision Number | Description |
|---|---|---|
| March 1986 | 0 | Original Issue |
| August 1986 | Revision 1 | Table 2-2 Corrected<br>Table 2-7 Corrected<br>Table 2-14 Corrected<br>Table 3-1 Corrected<br>Table 3-2 Corrected<br>Paragraph 4.4 New Paragraph<br>Paragraph 4.5 New Paragraph<br>Table 4-1 Updated<br>Table 4-2 Updated<br>Section 7 New Section<br>Appendix A New Appendix<br>Appendix B New Appendix<br>Minor rewrites and corrections were made throughout Revision 1 |

## List of Tables (Cont'd)



## List of Figures

# 1. INTRODUCTION

This report provides a review and update of the materials data and information for the reactor pressure vessels of North Anna Units 1 and 2 to ensure that they are in compliance with the requirements of 10CFR50, Appendix G.[1] In addition, the reactor pressure vessel surveillance programs were reviewed for compliance with 10CFR50, Appendix H.[2] The reactor pressure vessel surveillance capsule withdrawal schedule was modified to meet the intent of ASTM E185-82[3] as referenced by 10CFR50, Appendix H.

As a result of this review and update the reactor vessels materials data bases for North Anna Units 1 and 2 were found to be in compliance with 10CFR50, Appendix G. The surveillance program materials properties data bases are in compliance with 10CFR50, Appendix H and will provide the material data necessary to ensure continued licensibility of the reactor vessels.

A new reactor vessel surveillance capsule withdrawal schedule was developed to meet the requirements of ASTM E185-82 as referenced by 10CFR50, Appendix H. This new schedule will provide needed irradiation materials data in a timely manner.

The reason for this review and update is that the reactor pressure vessels were fabricated and the corresponding surveillance programs were developed prior to the implementation of 10CFR50, Appendixes G and H. These regulations recognized that the older plants could not meet all the requirements and established guidelines to meet the intent, if not the letter of the regulations. In addition, these regulations have been revised as experience, new data and analysis capability related to reactor vessel integrity have developed. A periodic review and update is necessary to ensure continued compliance with the regulations.

# 2. REACTOR VESSEL DATA BASES

The establishment of the mechanical and toughness properties of reactor pressure vessels in accordance with applicable regulations and standards is an essential aspect of the licensing process. As these rules are improved it is necessary to ensure that the data used for licensing of the reactor vessels are representative of the best information and materials properties available for each specific reactor vessel. The data are also essential in establishing the normal pressure-temperature operating limitations as required by 10CFR50, Appendix G.

## 2.1. North Anna Unit 1

The materials and chemical composition data for the North Anna Unit 1 reactor vessel are presented in Tables 2-1 through 2-6. These data represent an accumulation of information from various sources (References 4, 8 and 9) and are complete except for the limited amount of data for the weld metals. The current essential data for the weld metals are available. In addition, the initial reference temperature data represents the best available data as defined in Section 2.4.

The location and identification of the forgings and welds within the belt-line region of the North Anna Unit 1 reactor pressure vessel are shown in Figure 2-1.

## 2.2. North Anna Unit 2

The data for North Anna Unit 2 reactor vessel are presented in Tables 2-8 through 2-13. These data represent an accumulation of information from various sources (References 4, 10 and 11) and are complete except for the limited amount of data for the weld metals. The current essential data for the weld metals are available. In addition, the initial reference temperature data represents the best available data as defined in Section 2.4.

The location and identification of the forgings and welds within the beltline region of the North Anna Unit 2 reactor pressure vessel are shown in Figure 2-2.

## 2.3. Surveillance Data Bases

Each reactor vessel has a surveillance program to monitor the neutron radiation damage of the materials in the beltline region. These data for each reactor vessel at North Anna were tabulated separately from the main data base as a convenience for easy reference. The data are presented in Tables 2-7 and 2-14.

## 2.4. Initial Value of Reference Temperature

The initial value of reference temperature is not always available for the materials used to fabricate older reactor vessels because it was not an established requirement of the ASME Code. Even for reactor vessels completed after the establishment of the requirement, the value was often unattainable because no suitable material was available. The necessary drop weight test data were usually obtained for both plate and forging materials and this provided a reliable data base to establish the initial reference temperature for these materials.

The initial reference temperature of weld metals was not obtained until after it was required by the ASME Code. At that time an effort was made to re-evaluate the weld qualification to obtain initial reference temperatures. Subsequently, a statistical evaluation was made of weld metals fabricated after the establishment of the ASME Code requirement, plus reevaluation of old weld metals, to provide a basis for a mean value and standard deviation. A similar approach was used to establish values for plate and forging materials for which the needed actual test data were not available. An acceptable method for establishing the initial reference temperature is presented in the NRC Standard Review Plan, Section 5.3.2.[5]. It is recognized that the values recommended in the Standard Review Plan are very conservative.

Previous evaluations of the North Anna reactor vessels had established reference temperatures for the base materials which were representative of actual data. However, the weld metals had no data for the submerged-arc weldments made by Rotterdam. The initial value of Rotterdam welds were established per the NRC Standard Review Plan, Section 5.3.2.

A more recent development is the need to establish the standard deviation of the reactor vessel materials to be used in determining the reference temperature shift as a result of irradiation. Standard deviations for measured initial reference temperature have been established from data obtained since the ASME Code requirement for establishing the reference temperature of all reactor vessel materials.

Listed below are the initial reference temperatures and standard deviations used if actual measured values are not available:

| Material Description | Initial $RT_{NDT}$, F | $RT_{NDT}$ Std. Dev. | Reference |
|---|---|---|---|
| SA508,Cl. 2 | 3 | ±30F | BAW-10046P[6] |
| RDM Welds | 0 | ±20F | SECY 82-465[7] |

Table 2-1. Identification of Reactor Vessel Beltline Region Weld Metals - North Anna Unit-1

| Weld Identification | Weld Location | Welding Process | Weld Wire | | Flux | | Reference |
|---|---|---|---|---|---|---|---|
| | | | Type | Heat No. | Type | Lot No. | |
| W05A | Nozzle Shell to Interm. Shell Seam (O.D. 94%) | Sub. Arc | SMIT 40 | 25295 | SMIT 89 | 1170 | Docket No. 50-338[8] |
| W05B | Nozzle Shell to Interm. Shell Seam (I.D. 6%) | Sub. Arc | S4 Mo | 4278 | SMIT 89 | 1211 | " |
| W04# | Interm. Shell to Lower Shell Seam | Sub. Arc | SMIT 40 | 25531 | SMIT 89 | 1211 | " |

# - Denotes material included in Reactor Vessel Surveillance Program

Table 2-2. Chemical Composition of Reactor Vessel Beltline Region Weld Metals - North Anna Unit-1

| Material Identification | Chemical Composition, Weight Percent | | | | | | | | | | Reference |
|---|---|---|---|---|---|---|---|---|---|---|---|
| | C | Mn | P | S | Si | Cr | Ni | Mo | V | Cu | |
| W05A | 0.10 | 1.50 | [0.020] | — | 0.36 | — | [0.10] | 0.37 | — | 0.30 | Docket No. 50-338[8] |
| W05B | 0.09 | 1.49 | [0.020] | — | 0.33 | — | [0.10] | 0.37 | — | 0.11 | " |
| W04# | 0.06 | 1.29 | 0.020 | 0.012 | 0.35 | 0.25 | 0.11 | 0.49 | 0.001 | 0.086 | " |

# - Denotes material included in Reactor Vessel Surveillance Program

[ ] - Estimate based on review of similar material

Table 2-3. Mechanical Properties of Reactor Vessel Beltline Region Weld Metal - North Anna Unit 1

| Weld Identification | Toughness Properties | | | | Tensile Properties, RT | | | | Post Weld Heat Treatment | Reference |
|---|---|---|---|---|---|---|---|---|---|---|
| | $T_{NDT}$, F | 10F Energy, Ft-lbs | RT$_{NDT}$, F | USE, Ft-lbs | Strength, Ksi Yield | Ult. | Elong % | RA % | | |
| W05A | 0* | — | 0** | — | — | — | — | — | — | Docket No. 50-338(8) |
| W05B | 0* | — | 0** | — | — | — | — | — | — | * |
| W04† | -11 | — | 19 | 102 | — | — | — | — | — | * |

* - Estimated per NRC Standard Review Plan Section 5.3.2
** - Estimated per Section 2.4
† - Denotes material included in Reactor Vessel Surveillance Program

Table 2-1. Identification of Reactor Vessel Beltline Region Base Materials - North Anna Unit-1

| Material Identification Heat No. | Type | Component | Code No. | Supplier | Heat Treatment | Reference |
|---|---|---|---|---|---|---|
| 990286/ 295213 | SA508 CL.2 | Nozzle Shell Forging | 05 | Rheinstahl Huttenwerke | — | Docket No. 50-338(8) |
| 990311/ 298244 | SA508 CL.2 | Inter. Shell Forging | 04 | " | — | " |
| 990400/ 292332# | SA508 CL.2 | Lower Shell Forging | 03 | " | — | " |

# - Denotes material included in Reactor Vessel Surveillance Program

Table 2-5. Chemical Composition of Reactor Vessel Beltline Region Base Materials - North Anna Unit-1

| Material Identification | Chemical Composition, Weight Percent | | | | | | | | | | | Reference |
|---|---|---|---|---|---|---|---|---|---|---|---|---|
| | C | Mn | P | S | Si | Ni | Cr | Mo | Co | V | Cu | |
| 05 | 0.20 | 0.71 | 0.013 | 0.012 | 0.21 | 0.74 | 0.39 | 0.64 | — | 0.03 | 0.16 | Docket No. 50-338(8) |
| 04 | 0.21 | 0.75 | 0.010 | 0.019 | 0.21 | 0.82 | 0.33 | 0.64 | — | 0.05 | 0.12 | " |
| 03# | 0.19 | 0.68 | 0.009 | 0.014 | 0.22 | 0.80 | 0.30 | 0.63 | — | 0.02 | 0.15 | " |

# - Denotes material included in Reactor Vessel Surveillance Program

Table 2-6. Mechanical Properties of Reactor Vessel Beltline Region Base Materials - North Anna Unit 1

| Material Identification | Toughness Properties | | | Tensile Properties, RT | | | | Reference |
|---|---|---|---|---|---|---|---|---|
| | | | | Strength, Ksi | | | | |
| | $T_{NDT}$, F | $RT_{NDT}$, F | USE, Ft-Lbs | Yield | Ult. | Elong % | RA % | |
| 05 | 2 | 6* | -- | -- | -- | -- | -- | Docket No. 50-338[8] |
| 04 | -31 | 17 | 92 | -- | -- | -- | -- | " |
| 03# | -13 | 38 | 85 | -- | -- | -- | -- | " |

* - Estimated Per NRC Standard Review Plan Section 5.3.2.

# - Denotes material included in Reactor Vessel Surveillance Program

Table 2-7. Properties of Surveillance Program Plate and Weld Material - North Anna Unit-1

| Material Identification | Toughness Properties $T_{30ft}$, °F | Toughness Properties $RT_{30ft}$, °F | Toughness Properties USE, Ft-Lbs | Tensile Properties, RT Strength, Ksi Yield | Tensile Properties, RT Strength, Ksi Ult. | Tensile Properties, RT Elong % | Tensile Properties, RT RA % | Heat Treatment* | Reference |
|---|---|---|---|---|---|---|---|---|---|
| Forging-03 | -12 | +15 | 85 | 70.68 | 92.55 | 18.80 | 58.85 | A-1616-1725F-2-[illegible]/WQ<br>T-1202-1292F-7-[illegible]/FC<br>S-1130±25F-14-3/4HR/FC | WCAP-8771[9] |
| Weld Metal-W04 | -12 | -12 | 95 | 64.18 | 79.35 | 19.20 | 71.00 | S-1130±25F-10-3/4HR/FC | " |

| Material Identification | Chemical Composition, Weight Percent C | Mn | P | S | Si | Cr | Ni | Mo | Co | V | Cu | Reference |
|---|---|---|---|---|---|---|---|---|---|---|---|---|
| Forging-03 | 0.20 | 0.68 | 0.019 | 0.011 | 0.26 | 0.30 | 0.79 | 0.61 | 0.020 | 0.037 | 0.16 | WCAP-8771[9] |
| Weld Metal-W04 | 0.06 | 1.29 | 0.020 | 0.012 | 0.35 | 0.025 | 0.11 | 0.49 | 0.006 | 0.001 | 0.086 | " |

* - A = Austenitized/Water Quenched (WQ)
T = Tempered/Furnace Cooled (FC)
S = Stress Relieved/Furnace Cooled (FC)

Table 2-9. Identification of Reactor Vessel Beltline Region Weld Metals - North Anna Unit-2

| Weld Identification | Weld Location | Welding Process | Weld Wire Type | Weld Wire Heat No. | Flux Type | Flux Lot No. | Reference |
|---|---|---|---|---|---|---|---|
| W05A | Nozzle Shell to Interm. Shell Seam (O.D. 94%) | Sub. Arc | S4 Mo | 4278 | SMIT 89 | 1211 | Docket No. 50-339(10) |
| W05B | Nozzle Shell to Interm. Shell Seam (I.D. 6%) | Sub. Arc | S4 Mo | 801 | SMIT 89 | 1211 | " |
| W04# | Interm. Shell to Lower Shell Seam | Sub. Arc | S3 Mo | 716126 | LW320 | 26 | " |

# - Denotes material included in Reactor Vessel Surveillance Program

Table 2-9. Chemical Composition of Reactor Vessel Beltline Region Weld Metals - North Anna Unit-2

| Weld Identification | Chemical Composition, Weight Percent | | | | | | | | | | Reference |
|---|---|---|---|---|---|---|---|---|---|---|---|
| | C | Mn | P | S | Si | Cr | Ni | Mo | V | Cu | |
| W05A | 0.09 | 1.49 | — | — | 0.33 | — | [0.10] | 0.37 | — | 0.11 | Docket No. 50-339[10] |
| W05B | 0.086 | 1.58 | 0.012 | 0.012 | 0.43 | — | [0.10] | 0.51 | — | 0.18 | " |
| W04# | 0.08 | 1.82 | 0.017 | 0.011 | 0.25 | 0.042 | 0.084 | 0.49 | 0.002 | 0.088 | " |

# - Denotes material included in Reactor Vessel Surveillance Program

[ ] - Estimate based on review of similar material

Table 2-10. Mechanical Properties of Reactor Vessel Beltline Region Weld Metal - North Anna Unit 2

| Material Identification | Toughness Properties: $T_{NDT}$, °F | Toughness Properties: 10F Energy, Ft-lbs | Toughness Properties: $RT_{NDT}$, °F | Toughness Properties: USE, Ft-lbs | Tensile Properties, RT: Strength, Ksi Yield | Tensile Properties, RT: Strength, Ksi Ult. | Tensile Properties, RT: Elong, % | Tensile Properties, RT: RA % | Post Weld Heat Treatment | Reference |
|---|---|---|---|---|---|---|---|---|---|---|
| W05A | 0* | — | 0** | — | — | — | — | — | — | Docket No. 50-339[12] |
| W05B | 0* | — | 0** | — | — | — | — | — | — | " |
| W04# | -67 | — | -48 | 107 | — | — | — | — | — | " |

* - Estimated per NRC Standard Review Plan Section 5.3.2
** - Estimated per Section 2.4
# - Denotes material included in Reactor Vessel Surveillance Program

Table 2-11. Identification of Reactor Vessel Beltline Region Base Materials - North Anna Unit-2

| Material Identification Heat No. | Type | Component | Code No. | Supplier | Heat Treatment | Reference |
|---|---|---|---|---|---|---|
| 990598/ 291396 | SA508,Cl.2 | Nozzle Shell Forging | 05 | Rheinstahl Huttenwerke | — | Docket No. 50-339(10) |
| 990496/ 2924248 | SA508,Cl.2 | Inter. Shell Forging | 04 | " | — | " |
| 990533/ 297355 | SA508,Cl.2 | Lower Shell Forging | 03 | " | — | " |

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# - Denotes material included in Reactor Vessel Surveillance Program

Table 2-12. Chemical Composition of Reactor Vessel Beltline Region Base Materials - North Anna Unit-2

| Material Identification | Chemical Composition, Weight Percent | | | | | | | | | | | Reference |
|---|---|---|---|---|---|---|---|---|---|---|---|---|
| | C | Mn | P | S | Si | Ni | Cr | Mo | Co | V | Cu | |
| 05 | 0.20 | 0.68 | 0.010 | 0.013 | 0.25 | 0.77 | 0.34 | 0.60 | — | 0.01 | 0.08 | Docket No. 50-339[10] |
| 04# | 0.195 | 0.78 | 0.011 | 0.016 | 0.24 | 0.83 | 0.35 | 0.62 | — | 0.02 | 0.09 | " |
| 03 | 0.16 | 0.66 | 0.013 | 0.017 | 0.15 | 0.83 | 0.34 | 0.59 | — | 0.01 | 0.13 | " |

# - Denotes material included in Reactor Vessel Surveillance Program

Table 2-13. Mechanical Properties of Reactor Vessel Beltline Region Base Materials -North Anna Unit 2

| Material Identification | Toughness Properties | | | Tensile Properties, RT | | | | Reference |
|---|---|---|---|---|---|---|---|---|
| | | | | Strength, Ksi | | | | |
| | $T_{NDT}$, F | $RT_{NDT}$, F | USE, Ft-Lbs | Yield | Ult. | Elong % | RA % | |
| 05 | +5 | 9* | -- | -- | -- | -- | -- | Docket No. 50-339[10] |
| 04# | -49 | 75 | 74 | -- | -- | -- | -- | " |
| 03 | -13 | 56 | 80 | -- | -- | -- | -- | " |

* - Estimated Per NRC Standard Review Plan Section 5.3.2.

# - Denotes material included in Reactor Vessel Surveillance Program

Table 2-14. Properties of Surveillance Program Plate and Weld Material - North Anna Unit-2

| Material Identification | Toughness Properties $T_{NDT}$, °F | Toughness Properties $RT_{NDT}$, °F | Toughness Properties USE, Ft-lbs | Tensile Properties, RT Strength, ksi Yield | Tensile Properties, RT Strength, ksi Ult. | Tensile Properties, RT Elong % | Tensile Properties, RT RA % | Heat Treatment* | Reference |
|---|---|---|---|---|---|---|---|---|---|
| Forging-04 | -48 | +57 | 75 | 84.90 | 102.00 | 19.00 | 48.20 | A-1688-1697F-2 1/2HR/WQ<br>T-1220-1229F-6HR/FC<br>S-1130±25F-14 3/4HR/FC | WCAP-8772(11) |
| Weld Metal-W04 | -66 | -52 | 115 | 76.08 | 85.90 | 24.20 | 69.25 | S-1130±25F-13 1/2HR/FC | " |

| Material Identification | Chemical Composition, Weight Percent C | Mn | P | S | Si | Cr | Ni | Mo | Co | V | Cu | Reference |
|---|---|---|---|---|---|---|---|---|---|---|---|---|
| Forging-04 | 0.19 | 0.76 | 0.018 | 0.011 | 0.25 | 0.35 | 0.86 | 0.60 | 0.003 | 0.031 | 0.11 | WCAP-8772(11) |
| Weld Metal-W04 | 0.08 | 1.82 | 0.017 | 0.011 | 0.25 | 0.042 | 0.084 | 0.49 | <0.002 | 0.002 | 0.088 | " |

* - A = Austenitized/Water Quenched (WQ)
T = Tempered/Furnace Cooled (FC)
S = Stress Relieved/Furnace Cooled (FC)

Figure 2-1. Location and Identification of Materials Used in the Fabrication of the Belt-Line Region of North Anna Unit-1 Reactor Pressure Vessel (Ref. 12)

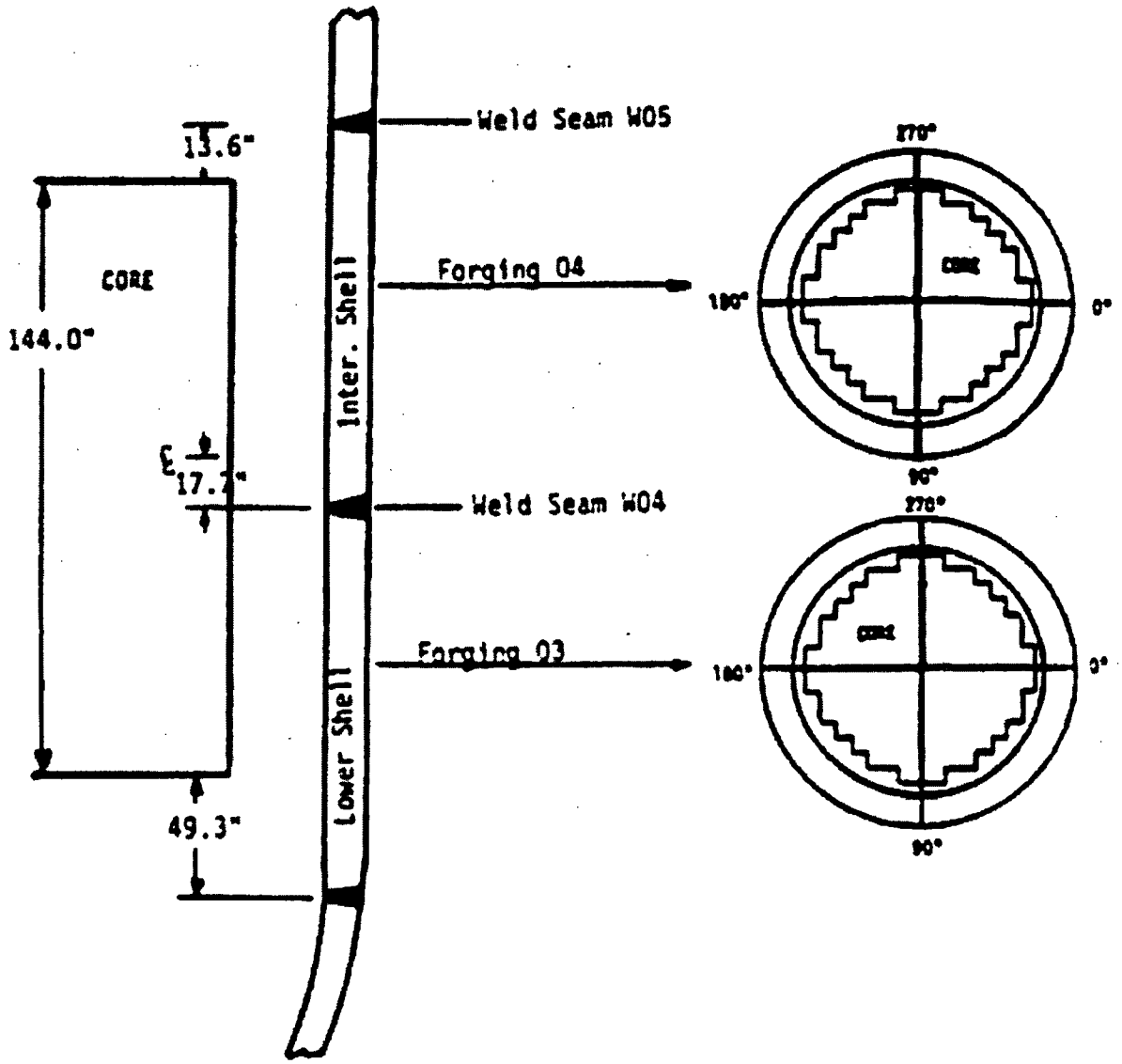
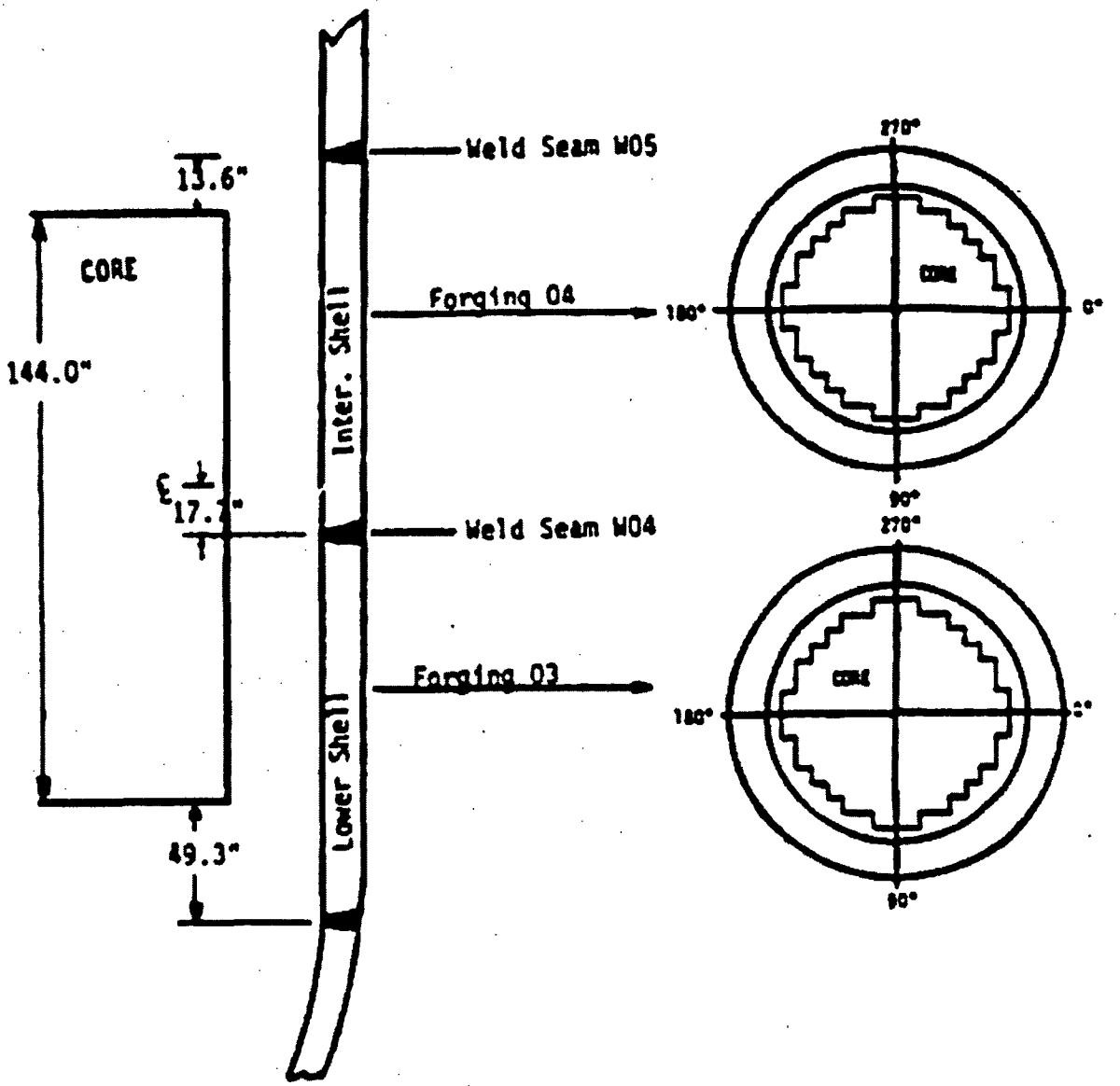


Figure 2-2. Location and Identification of Materials Used in the Fabrication of the Belt-Line Region of North Anna Unit-2 Reactor Pressure Vessel (Ref. 12)



# 3. EVALUATION OF REACTOR VESSEL TOUGHNESS

One aspect of reactor pressure vessel licensibility is the toughness of the materials used in its fabrication. These properties are used to calculate the pressure-temperature operating limits in accordance with the requirements of 10CFR50, Appendix G. The objective of these limits is to prevent nonductile failure of the reactor vessel during any normal operating condition, including anticipated operational occurrences and system hydrostatic tests.

The closure head region, the reactor vessel outlet nozzle, and the beltline region have been identified as the only regions of the reactor vessel that regulate the pressure-temperature limits. Since the closure head region is significantly stressed at relatively low temperatures (due to mechanical loads resulting from bolt preload), this region largely controls the pressure-temperature limits of the first several service periods. The reactor vessel outlet nozzle also affects the pressure-temperature limit curves of the first several service periods. This is due to the high local stresses at the inside corner of the nozzle, which can be two to three times the membrane stresses of the shell. After the first several years of neutron radiation exposure, the $RT_{NDT}$ of the beltline region materials will be high enough that the beltline region of the reactor vessel will start to control the pressure-temperature limits of the reactor coolant pressure boundary (RCP). For the service period for which the limit curves are established, the maximum allowable pressure as a function of fluid temperature is obtained through the point-by-point comparison of the limits imposed by the closure head region, the outlet nozzle, and the beltline region. The maximum allowable pressure is taken to be the lowest of the three calculated pressures.

The unirradiated toughness properties of each reactor pressure vessel were determined for the belt-line region materials in accordance with 10CFR50, Appendix G. For the other belt-line region materials for which the measured properties are not available, the unirradiated impact properties and residual elements, as originally established for the belt-line region materials, were determined from acceptable data bases using recognized estimating techniques. The adjusted reference temperatures are calculated by adding the predicted radiation-induced shift of the $RT_{NDT}$ to the unirradiated $RT_{NDT}$ including margin. The predicted shift, $RT_{NDT}$, is calculated using the respective neutron fluence, plus the copper and nickel contents. The neutron fluence values are calculated based on function derived from adjoint functions using transport calculations. The design curves of regulatory Guide 1.99, Rev. 2,[13] were used to predict the radiation-induced shift of $RT_{NDT}$.

The results of the evaluation are presented in Tables 3-1 and 3-2 which show that both reactor pressure vessels have $RT_{NDT}$ values which will permit normal operation to the expiration of current licenses.

## Table 3-1. Evaluation of Reactor Pressure Vessel Fracture Toughness - North Anna Unit-1

| Material Identification Heat No. | Type | Beltline Region Location | Chemical Composition[a] Copper w/o | Nickel w/o | Initial $RT_{NDT}$, °F[a] | Margin, °F[c] | Fluence at Current License Expiration[b] Inside Surface, n/cm² | T/4 Location, n/cm² | $RT_{NDT}$ at Current License Expiration[c] Inside Surface, °F | T/4 Location, °F |
|---|---|---|---|---|---|---|---|---|---|---|
| Forging 05 | SA508,Cl.2 | Nozzle Shell Forging | 0.16 | 0.74 | 6 | 34 | 2.49E18 | 1.46E18 | 116 | 106 |
| Forging 04 | SA508,Cl.2 | Interm. Shell Forging | 0.12 | 0.62 | 17 | 34 | 3.11E19 | 1.81E19 | 163 | 149 |
| Forging 03 | SA508,Cl.2 | Lower Shell Forging | 0.16 | 0.80 | 38 | 34 | 3.11E19 | 1.81E19 | 232 | 212 |
| Weld 05A | Subm. Arc | Noz. Int. Cir. Weld | 0.30 | 0.10 | 0 | 69 | — | 1.46E18 | — | 166 |
| Weld 05B | Subm. Arc | Noz. to Int. Cir. Weld | 0.11 | 0.10 | 0 | 69 | 2.49E18 | — | 105 | — |
| Weld 04 | Subm. Arc | Lower Cir. Weld | 0.086 | 0.11 | 19 | 56 | 3.11E19 | 1.81E19 | 142 | 133 |

[a] Per Tables 2-1 through 2-4
[b] Per WCAP-11016[12]
[c] Per Regulatory Guide 1.99, Rev. 2[13]

## Table 3-2. Evaluation of Reactor Pressure Vessel Fracture Toughness - North Anna Unit-2

| Material Identification Heat No. | Type | Beltline Region Location | Chemical Composition[a] Copper w/o | Nickel w/o | Initial $RT_{NDT}$, F[a] | Margin, F[c] | Fluence at Current License Expiration[b] Inside Surface, n/cm² | T/4 Location, n/cm² | $RT_{NDT}$ at Current License Expiration[c] Inside Surface, F | T/4 Location, F |
|---|---|---|---|---|---|---|---|---|---|---|
| Forging 05 | SA508,Cl.2 | Nozzle Shell | 0.08 | 0.77 | 9 | 34 | 2.72E18 | 1.60E18 | 76 | 72 |
| Forging 04 | SA508,Cl.2 | Intermm. Shell | 0.09 | 0.83 | 75 | 34 | 3.40E19 | 2.00E19 | 186 | 177 |
| Forging 03 | SA508,Cl.2 | Lower Shell | 0.13 | 0.83 | 56 | 34 | 3.40E19 | 2.00E19 | 217 | 201 |
| Weld 05A | Subm. Arc | Noz. to Int. Cir. Weld | 0.11 | 0.10 | 0 | 69 | — | 1.60E18 | — | 102 |
| Weld 05B | Subm. Arc | Noz. to Int. Cir. Weld | 0.18 | 0.10 | 0 | 69 | 2.72E18 | — | 125 | — |
| Weld 04 | Subm. Arc | Int. to Low. Cir. Weld | 0.088 | 0.084 | -48 | 56 | 3.40E19 | 2.00E19 | 74 | 57 |

(a) Per Tables 2-8 through 2-13
(b) Per WCAP-11016[12]
(c) Per Regulatory Guide 1.99, Rev. 2[13]

# 4. REACTOR VESSEL SURVEILLANCE PROGRAMS

The design of a reactor vessel materials surveillance program is based on the need to monitor the toughness properties of the controlling radiation sensitive material from which the reactor vessel was fabricated. Of equal importance is the benchmarking, or verification, of the fluence which the reactor vessel experiences.

The extent to which a surveillance program meets these objectives depends on when the reactor vessel was fabricated. This is due to the evolution of surveillance program requirements as more knowledge has been obtained from existing programs. Some of the requirements can be upgraded to meet the current 10CFR50, Appendix H while other reactor vessels will be required to make do with the installed programs. Each of the North Anna surveillance programs will be described separately.

## 4.1. North Anna Unit 1

The surveillance program was designed in accordance with 10CFR50, Appendix H, and ASTM E185-73. The controlling materials are contained in the program. The program is updated by establishing a new withdrawal schedule which was developed around the current requirements as defined in ASTM E185-82 and the desire to move capsules from low lead factor sites to high lead factor sites only during scheduled ten year reactor vessel inspections.

Two methods are permitted to determine when a capsule is to be removed for evaluation; i.e., EFPY exposure or cumulative fluence. The cumulative fluence was used to establish the new schedule which will match the materials data obtained at each capsule evaluation to the critical times in the reactor vessel design life. The EFPY schedule would produce results

with too low cumulative fluence to provide useful irradiation materials data.

The new proposed withdrawal schedule for North Anna Unit 1 is shown in Table 4-1.

## 4.2. North Anna Unit 2

The surveillance program was designed in accordance with 10CFR50, Appendix H, and ASTM E185-73. The controlling materials are contained in the program. The program is updated by establishing a new withdrawal schedule based on accumulative fluence in accordance with ASTM E185-82. The new proposed withdrawal schedule for North Anna Unit 2 is shown in Table 4-2.

## 4.3. Spare Capsules

Extra Surveillance Capsules not required to meet the current requirements of ASTM E185-82 will remain in the reactor vessel. These capsules can be used in the future to provide data for the verification of reactor vessel fluence calculations or to provide materials data for support of plant life extension. To ensure that the extra capsules will provide useful data in the future they will be moved to maximum lead factor positions during the appropriate inservice inspection as the positions become available, in order to maximize the total accumulated fluence.

## 4.4. Archive Surveillance Material

The availability of additional surveillance material is important in the event that additional materials properties data is needed for the reactor vessel controlling beltline region materials; or, in the event that licensing requirements change and new data is required which necessitates applying improved testing techniques to the original materials of fabrication. An inventory of the available materials was made for North Anna Units 1 and 2. The detailed descriptions of the inventory are presented in Appendix A.

## 4.5. Capsule Evaluation Results

The results of the evaluation of the surveillance capsules removed from North Anna 1 and 2 are summarized in Appendix B. The capsules removed to date are:

North Anna Unit 1 - Capsule V

North Anna Unit 2 - Capsule V

Table 4-1. Revised Surveillance Capsule Withdrawal Schedule - North Anna Unit-1

1 February 1986

| Recommended Capsule Withdrawal Schedule Per 10CFR50, Appendix H, and E185-82 | | | Revised Withdrawal Schedule* | | | | |
|---|---|---|---|---|---|---|---|
| Capsule Sequence | Vessel EFPY | Capsule Fluence, n/cm$^2$ | Capsule I.D. | EFPY | Estimated Capsule[12] fluence, n/cm$^2$ | Estimated Maximum[12] RV fluence, n/cm$^2$ | Estimated Withdrawal, Yr |
| First | 1.5 | 5E18 or $RT_{NDT}$>50F | V** | 1.1 | 3.13E18 | 1.89E18 | — |
| Second | 3 | 9E18 | U | 5.8 | 9.02E18 | 8.61E18 | 1987 |
| Third | 6 | 1.8E19 (EOL T/4) | X | 8.1 | 1.88E19 | 1.13E19 | 1990 |
| Fourth | 15 | 3.1E19 (EOL I.W.) | W*** | 16.9 | 3.05E19 | 2.16E19 | 2002 |
| Fifth | EOL | >3.1E19 | Y**** | 25 | 4.21E19 | 3.11E19 | 2011 |

NOTE: Remainder of capsules (Z, T, and S) moved to maximum lead factor position during inservice inspections as positions become available in order to maximize total fluence.

---

* - Reviewed after each fuel cycle and revised as needed after each capsule withdrawal and evaluation. Estimated withdrawals based on 18-month fuel cycles and 0.80 plant capacity.

** - Capsule withdrawn and evaluated.

*** - Transferred from 0.82 lead factor location to 1.52 lead factor location during the 10 year inservice inspection (May 1987/5.89 EFPY).

**** - Transferred from 0.82 lead factor location to 1.52 lead factor location during the 20 year inservice inspection (May 1997/13.58 EFPY)

Table 4-2. Revised Surveillance Capsule Withdrawal Schedule - North Anna Unit-2

1 February 1986

| Recommended Capsule Withdrawal Schedule Per 10CFR50, Appendix H, and E185-82 | | | Revised Withdrawal Schedule* | | | | |
|---|---|---|---|---|---|---|---|
| Capsule Sequence | Vessel EFPY | Capsule Fluence, n/cm$^2$ | Capsule I.D. | EFPY | Estimated Capsule[12] fluence, n/cm$^2$ | Estimated Maximum[12] RV fluence, n/cm$^2$ | Estimated Withdrawal, Yr |
| First | 1.5 | 5E18 or $RT_{NDT}$>50F | V** | 1.0 | 2.87E18 | 1.73E18 | — |
| Second | 3 | 1.0E19 | U | 6.1 | 1.03E19 | 8.95E18 | 1989 |
| Third | 6 | 2.0E19 (EOL T/4) | X | 7.1 | 1.78E19 | 1.04E19 | 1990 |
| Fourth | 15 | 3.4E19 (EOL I.W.) | W*** | 17.0 | 3.60E19 | 2.43E19 | 2004 |
| Fifth | EOL | >3.4E19 | Y*** | 23.8 | 5.25E19 | 3.40E19 | 2011 |

NOTE: Remainder of capsules (Z, T, and S) moved to maximum lead factor position during inservice inspections as positions become available in order to maximize total fluence.

* - Reviewed after each fuel cycle and revised as needed after each capsule withdrawal and evaluation. Estimated withdrawals based on 18-month fuel cycles and 0.80 plant capacity.

** - Capsule withdrawn and evaluated.

*** - Transferred from 0.82 lead factor location to 1.52 lead factor location during the 10 year inservice inspection (August 1990/7.15 EFPY).

## 5. INTEGRATED REACTOR VESSEL SURVEILLANCE PROGRAM

The idea of an integrated reactor vessel surveillance program develops whenever two or more nuclear plants share the same site, or when owned by the same utility and share a common design. It is readily apparent that a savings can be recognized from reduced capsule evaluations and, of special importance, reduced worker radiation exposure. The requirements set forth in 10CFR50, Appendix H, as applicable to integrated surveillance programs are as follows:

> "C. An integrated surveillance program may be considered for a set of reactors that have similar design and operating features. The representative materials chosen for surveillance from each reactor in the set may be irradiated in one or more of the reactors, but there must be an adequate dosimetry program for each reactor. No reduction in the requirements for number of materials to be irradiated, specimen types, or number of specimens per reactor is permitted, but the amount of testing may be reduced if the initial results agree with predictions. Integrated surveillance program must be approved by the Director, Office of Nuclear Reactor Regulation, on a case-by-case basis. Criteria for approval include the following considerations:
>
> 1. The design and operating features of the reactor in the set must be sufficiently similar to permit accurate comparisons of the predicted amount of radiation damage as a function of total power output.
>
> 2. There must be adequate arrangement for data sharing between plants.
>
> 3. There must be a contingency plan to assure that the surveillance program for each reactor will not be jeopardized by operation at reduced power level or by an extended outage of another reactor from which data are expected.

4. There must be a substantial advantage to be gained, such as reduced power outages or reduced personnel exposure to radiation, as a result of not requiring surveillance capsules in all reactors in the set."

The four requirements as set forth above can be met by the Virginia Power plants. The plants are similar in design and power ratings, especially as to current knowledge as to flux rate effects on neutron radiation damage. Since the plants have the same owner no problem would exist as to data sharing. Each plant could continue to have a plant specific surveillance program as a backup in case the other plants experienced a protracted shutdown. Finally, there would be a gain from the reduced personnel exposed to radiation.

The fault of this approach is the wording in the first paragraph, "... must be an adequate dosimetry program for each reactor". In recent years, the monitoring of the neutron fluence the reactor vessel receives has developed to be of equal importance with monitoring material damage. In fact, because of the large uncertainties that can be assigned to fluence analysis, if not properly verified (i.e., dosimeters in removed surveillance capsules), when combined with materials property changes can produce restricted pressure-temperature operations which could more than offset savings to be realized from a surveillance capsule evaluation. On the other hand, if a capsule is removed to benchmark a fluence determination a major portion of the cost is associated with the dosimeter and fluence evaluation. Therefore, it would be more practical to perform the complete capsule evaluation.

In addition, the new revision of Regulatory Guide 1.99, gives credit for obtaining surveillance data for the controlling materials. When two or more credible data points become available from a reactor they may be used to determine the adjusted reference temperature and decrease in Charpy upper shelf energy. With the exception of Surry Unit 1, all the plants have the controlling materials in their surveillance programs. Thus, the data should provide the best evaluation of

material damage to minimize the effect on operating limitations. In the case of North Anna Unit 1, the weld metal in the surveillance program is similar to the controlling weld metal and may be used to provide a high degree of confidence that prediction techniques are not unduly restricting the operating limits.

In summary, while an integrated reactor vessel surveillance program for North Anna Units 1 and 2 may be acceptable from a regulatory viewpoint, it would not be practical, since capsules would have to be withdrawn from each unit in order to provide a fluence benchmark. However, this option may become practical in the future and should be re-evaluated after additional capsules have been removed and evaluated.

## 6. SUMMARY

As a result of this review and update the reactor vessels materials data bases for North Anna Units 1 and 2 were found to be in compliance with 10CFR50, Appendix G. The surveillance program materials properties data bases and capsule withdrawal schedules are in compliance with 10CFR50, Appendices G and H to the extent practical and will provide the material data necessary to insure continued compliance with these appendices.

## 7. CERTIFICATION

The reactor pressure vessel material property data bases and the reactor vessel material surveillance programs for Virginia Electric and Power Company North Anna Units 1 and 2 were updated in accordance with the requirements of 10 CFR 50, Appendixes G and H.

A. L. Lowe, Jr., P. E. — Date 9/3/86
Project Technical Manager

This report has been reviewed for technical content and accuracy.

L. B. Gross, P. E. — Date 9/3/86
Principal Engineer

This report has been approved for release.

D. J. Firth — Date 9/3
Project Manager

## 8. REFERENCES


1. U.S. Code of Federal Regulations, Title 10, Energy, Part 50, "Domestic Licensing of Production and Utilization Facilities, Appendix G, Facture Toughness Requirements."

2. U.S. Code of Federal Regulations, Title 10, Energy, Part 50, "Domestic Licensing of Production and Utilization Facilities, Appendix H, Reactor Vessel Material Surveillance Program Requirements."

3. ASTM Standard E185-82, "Practice for Conducting Surveillance Tests for Light Water-Cooled Nuclear Power Reactor Vessels," ASTM Standards 03.01, August 1985.

4. North Anna Power Station Units 1 and 2, Updated Final Safety Analysis Report, Virginia Electric and Power Company, July 20, 1982, as amended.

5. United States Nuclear Regulatory Commission, Standard Review Plan Branch Technical Position 5-2, Revision 1 NUREG-0800, July 1981.

6. H. S. Palme, et al., "Methods of Compliance With Fracture Toughness and Operational Requirements of 10CFR50, Appendix G," BAW-10046P, Babcock & Wilcox, Lynchburg, Virginia, March 1976.

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

8. Letter from C. M. Stallings, Virginia Electric and Power Company to Harold R. Denton, Office of Nuclear Reactor Regulation, Subject: 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.

9. 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.

10. Letter from C. M. Stallings, Virginia Electric and Power Company to Harold R. Denton, Office of Nuclear Reactor Regulation, Subject: Pressure Vessel Fracture Toughness Properties, North Anna Power Station, Unit Nos. 1 and 2, Docket No. 50-339, December 11, 1978, Public Document Accession No. 7812150277.

11. J. A. Davidson, et al., "Virginia Electric and Power Company North Anna Unit No. 2 Reactor Vessel Radiation Surveillance Program," WCAP-8772, Westinghouse Electric Corporation, Pittsburgh, Pennsylvania, November 1976.

12. E. L. Furchi, et al., "North Anna Units 1 and 2 Reactor Vessel Fluence and $RT_{PTS}$ Evaluations" WCAP-11016 Revision 1, Westinghouse Electric Corporation, Pittsburgh, Pennsylvania, December 1985.

13. "Effect of Residual Elements on Predicted Radiation Damage to Reactor Vessels," U. S. NRC Regulatory Guide 1.99, Revision 2, Draft dated August 14, 1985.

# APPENDIX A

## ARCHIVE SURVEILLANCE MATERIAL

## ARCHIVE SURVEILLANCE MATERIAL

An inventory of the archive material available from the reactor vessel surveillance programs of the North Anna Units 1 and 2 showed that the following materials are in storage at the reactor system supplier.

North Anna Unit 1:

No archive surveillance material for this reactor pressure vessel was located at the reactor system supplier.

North Anna Unit 2:

Intermediate shell forging/forging No. 990496/292424;
1 piece - 3" x 9-3/4" x 19"
Weld metal W04 (Forging 990496/292424 to forging 990533/297355);
1 piece - 3" x 6-1/2" x 6-1/2"

# APPENDIX B

## SUMMARY OF SURVEILLANCE CAPSULE RESULTS

## North Anna Unit 1 - Capsule V

Capsule fluence: $3.13 \times 10^{18} n/cm^2$ (12)

Tension Test Results (Unirradiated):

| Specimen number | Test temp. F | Strength, psi 0.2 yield | Strength, psi Ultimate | Elongation, % Unif. | Elongation, % Total | R.in A. % |
|---|---|---|---|---|---|---|
| **Base Metal (990400/292332) Axial** | | | | | | |
| -- | RT | 70,700 | 92,500 | 11.9 | 18.8 | 59 |
| -- | 300 | 64,400 | 84,900 | 13.4 | 22.6 | 63 |
| -- | 600 | 55,600 | 86,200 | 15.5 | 23.3 | 55 |
| **Weld Metal (W-04)** | | | | | | |
| -- | RT | 64,200 | 79,400 | 9.7 | 19.2 | 71 |
| -- | 300 | 62,000 | 75,800 | 9.6 | 21.0 | 68 |
| -- | 600 | 60,900 | 78,700 | 9.4 | 19.0 | 60 |

Tension Test Results (Irradiated)

| Specimen number | Test temp. F | Strength, psi 0.2 yield | Strength, psi Ultimate | Elongation, % Unif. | Elongation, % Total | R.in A. % |
|---|---|---|---|---|---|---|
| **Base Metal (990400/292332) Axial** | | | | | | |
| VT-3 | RT | 80,400 | 99,300 | 12.0 | 20.9 | 55 |
| VT-4 | 548 | 64,700 | 95,700 | 10.0 | 18.4 | 47 |
| **Weld Metal** | | | | | | |
| VW-4 | RT | 70,800 | 84,500 | 13.0 | 19.3 | 65 |
| VW-3 | 548 | 63,600 | 84,500 | 11.7 | 19.0 | 57 |

Charpy Impact Test Results

Increase in 30 ft-lb Transition Temperature, F:

| Material | Unirradiated | Irradiated | Shift | Predicted R.G.1.99/2 |
|---|---|---|---|---|
| Base Metal (Axial) | +46 | +65 | 19 | 84 |
| Base Metal (Tangent.) | -6 | +33 | 39 | 84 |
| Heat-Affected-Zone | -51 | +4 | 55 | 84 |
| Weld Metal | -26 | +52 | 78 | 35 |

Decrease in Upper Shelf Energy, ft-lbs.

| Material | Unirradiated | Irradiated | Shift | Predicted R.G.1.99/2 |
|---|---|---|---|---|
| Base Metal (Axial) | 85 | 75 | 10 | 16 |
| Base Metal (Tangent.) | 135 | 122 | 13 | 26 |
| Heat-Affected-Zone | 145 | 105 | 40 | 28 |
| Weld Metal | 95 | 95 | 0 | 16 |

Reference: A. L. Lowe, Jr., et al, "Analysis of Capsule V, Virginia Electric & Power Company North Anna Unit No. 1 Reactor Vessel Materials Surveillance Program," BAW-1638, Babcock & Wilcox, Lynchburg, Virginia, May 1981

## North Anna Unit 2 - Capsule Y

Capsule fluence: $2.87 \times 10^{18} n/cm^2$ (12)

Tension Test Results (Unirradiated):

| Specimen number | Test temp. F | Strength, psi 0.2 yield | Strength, psi Ultimate | Elongation, % Unif. | Elongation, % Total | R.in A. % |
|---|---|---|---|---|---|---|
| **Base Metal (990496/292424) Axial** | | | | | | |
| -- | RT | 84,900 | 102,000 | 13.2 | 19.0 | 48 |
| -- | 300 | 77,700 | 95,200 | 10.5 | 16.7 | 50 |
| -- | 550 | 75,800 | 97,200 | 11.5 | 18.1 | 48 |
| **Weld Metal** | | | | | | |
| -- | RT | 76,100 | 86,000 | 14.2 | 24.2 | 69 |
| -- | 300 | 68,500 | 77,100 | 10.7 | 20.9 | 67 |
| -- | 550 | 63,900 | 80,100 | 11.9 | 21.5 | 62 |

Tension Test Results (Irradiated)

| Specimen number | Test temp. F | Strength, psi 0.2 yield | Strength, psi Ultimate | Elongation, % Unif. | Elongation, % Total | R.in A. % |
|---|---|---|---|---|---|---|
| **Base Metal (990496/292424) Axial** | | | | | | |
| GT-4 | 68 | 85,600 | 104,500 | 7.5 | 18.4 | 45 |
| GT-3 | 550 | 90,100 | 98,700 | 5.1 | 13.3 | 51 |
| **Weld Metal** | | | | | | |
| GW-3 | 68 | 78,600 | 91,600 | 6.2 | 19.7 | 64 |
| GW-4 | Specimen damaged during equipment malfunction | | | | | |

Charpy Impact Test Results

Increase in 30 ft-lb Transition Temperature, F:

| Material | Unirradiated | Irradiated | Shift | Predicted R.G.1.99/2 |
|---|---|---|---|---|
| Base Metal (Axial) | +60 | +69 | 9 | 51 |
| Base Metal (Tangent.) | -13 | -4 | 9 | 51 |
| Heat-Affected-Zone | -54 | -44 | 10 | 51 |
| Weld Metal | -26 | -24 | 2 | 32 |

Decrease in Upper Shelf Energy, ft-lbs.

| Material | Unirradiated | Irradiated | Shift | Predicted R.G.1.99/2 |
|---|---|---|---|---|
| Base Metal (Axial) | 75 | 62 | 13 | 11 |
| Base Metal (Tangent.) | 120 | 120 | 0 | 18 |
| Heat-Affected-Zone | 95 | 89 | 6 | 16 |
| Weld Metal | 115 | 92 | 23 | 20 |

Reference: A. L. Lowe, Jr., et al, "Analysis of Capsule V, Virginia Electric & Power Company Reactor Vessel Materials Surveillance Program," BAW-1794, Babcock & Wilcox, Lynchburg, Virginia, October 1983.