BAW-2224
JULY 1994

NORTH ANNA UNITS 1 AND 2 RESPONSE TO CLOSURE LETTER FOR NRC GENERIC LETTER 92-01, REVISION 1

B&W NUCLEAR TECHNOLOGIES

BAW-2224
July 1994

NORTH ANNA UNITS 1 AND 2
RESPONSE TO CLOSURE LETTER FOR
NRC GENERIC LETTER 92-01, REVISION 1

by

M. J. DeVan

Prepared for

Virginia Power

BWNT Document No. 77-2224-00
(See Section 7 for document signatures)

Prepared by

B&W NUCLEAR TECHNOLOGIES, INC.
Engineering and Project Services Division
P. O. Box 10935
Lynchburg, Virginia 24506-0935

B&W NUCLEAR TECHNOLOGIES

CONTENTS

List of Tables

1. INTRODUCTION

This report provides a response to the Generic Letter 92-01, Revision 1, closure letter recently issued by the U. S. Nuclear Regulatory Commission for Virginia Power's North Anna Unit 1 (NA1) and North Anna Unit 2 (NA2) Nuclear Station.

2. ORGANIZATION OF RESPONSE

The Generic Letter closure letter requested information which included the following:

1. For nclosure 1, "Data Summary for Pressurized Thermal Shock," verify that the information therein is accurate.

2. For Enclosure 2, "Data Summary for Upper-Shelf Energy," verify that the information therein is accurate.

3. Additional information.

    (a) Upper-shelf energy for forgings

    (b) Nickel content for Rotterdam weld.

Expanded explanations for the additional information are presented in following Sections 3 and 4.

In Section 5 the full Data Summary Tables for Pressurized Thermal Shock and Upper-Shelf Energy are presented. Those values that are <u>unchanged</u> are shown in <u>shaded</u> boxes. Where applicable, the table presentation was reordered from the Generic Letter 92-01 closure letter enclosures supplied by the Nuclear Regulatory Commission (NRC). The order of presentation followed in this report is at follows:

1. Wrought materials (forgings and plate) arranged from top to bottom as located in the reactor vessel

2. Circular welds arranged from top to bottom as located in the reactor vessel

The NRC's closure letter for NA1 and NA2 is shown following the above information.

3. UPPER-SHELF ENERGY FOR NOZZLE BELT FORGING

The North Anna (NA) Updated Final Safety Analysis Report[1] (UFSAR) Table 5.2-26 indicates a Charpy absorbed energy value of 60 ft-lbs (tangential, unirradiated) for the reactor vessel nozzle belt Forging 05 at a test temperature of ≤68°F. (This upper-shelf data is repeated in WCAP-11791,[2] Table III.) If this test data were assumed to characterize the upper-shelf of this material, the initial Charpy V-notch upper-shelf energy (CvUSE) of this material in the axial orientation might be conservatively estimated by application of the guidelines in the Standard Review Plan, Branch Technical Position MTEB 5-2,[3] "Fracture Toughness Requirements," for estimating the CvUSE which requires the tangential CvUSE values to be multiplied by 0.65. (See also Memo from C. Z. Serpan, Jr. to C. Y. Cheng, "Ratio of Transverse to Longitudinal Orientation Charpy Upper-Shelf Energy," dated June 25, 1990.) Applying this method would result in an estimated initial CvUSE of 39 ft-lbs in the axial orientation. However, the 60 ft-lbs unirradiated Charpy absorbed energy value for Forging 05 in the tangential orientation is not representative of the material strength at upper-shelf conditions. Further, there is evidence which supports the conclusion that the initial CvUSE in the axial orientation for the NA1 Forging 05 is 74 ft-lbs or more. The basis for this conclusion is presented in the following paragraphs.

An unirradiated CvUSE estimate of 74 ft-lbs for Forging 05 is supported by consideration of the data in Table 5.2-26 of NA UFSAR (or Table III of WCAP-11791) and Appendix C of BAW-1638,[4] "Analysis of Capsule V Virginia Electric and Power Company North Anna Unit No. 1 Reactor Vessel Materials Surveillance Program." Figure C-2 of BAW-1638 presents Charpy curves which demonstrate an average CvUSE of 135 ft-lbs and a minimum CvUSE value of 120 ft-lbs in the tangential orientation for the Forging 03 surveillance material. The NA UFSAR Table 5.2-26 (or Table III of WCAP-11791) presents an absorbed energy value of 74 ft-lbs in the tangential orientation for Forging 03 and a comparable absorbed

energy value of 60 ft-lbs in the tangential orientation for Forging 05. The values presented in Table 5.2-26 of the NA UFSAR are the minimum absorbed energy values at the highest test temperature (≤68°F). It is evident by inspection of the surveillance data for Forging 03 that the maximum test temperature (68°F) is considerably below the upper-shelf, since the tangentially oriented specimen data at 68°F (i.e., 74 ft-lbs) is less than the axially oriented specimen data at the upper-shelf (i.e., 85 ft-lbs). Similarly, it is evident that a 60 ft-lb absorbed energy value for a tangentially oriented specimen tested at 68°F is not consistent with the value which would be obtained for an axially oriented specimen tested at the upper-shelf, and certainly not for a tangentially oriented specimen tested at the upper-shelf.

Given the common vendor and nearly identical chemical compositions of Forgings 03, 04, and 05, the axially oriented CvUSE for either Forging 03 (85 ft-lbs) or Forging 04 (92 ft-lbs) would serve as an appropriate surrogate for the unirradiated CvUSE for Forging 05 in the axial orientation. As described in a Virginia Electric and Power Company letter to the NRC dated December 29, 1992,[5] the CvUSE for NA2 Forging 05 in the axial orientation was equated to that of the NA2 surveillance Forging 04 (i.e., 74 ft-lbs) in the axial orientation on the basis of a similar, quantitative comparison of tangential Charpy test data taken below 68°F. The NA2 surveillance Forging 04 CvUSE value in the axial orientation (i.e., 74 ft-lbs) is coincidentally equal to the NA1 surveillance Forging 03 measured Charpy data taken at a test temperature below 68°F in the tangential orientation. Based on this evidence, it can be concluded that the unirradiated CvUSE for the NA1 Forging 05 in the axial orientation is greater than the unirradiated CvUSE for the NA1 surveillance Forging 03 material in the tangential orientation at test temperature below 68°F. Therefore, a conservative estimate for the unirradiated CvUSE for Forging 05 in the axial orientation is 74 ft-lbs.

Because no known Rheinstahl Huttenwerke reactor vessel forgings are considered "low upper-shelf energy materials" (i.e., initial upper-shelf energy less than the minimum 75 ft-lbs required by 10CFR50 Appendix G[6] at the time of fabrication), it is concluded that an initial CvUSE of 74 ft-lbs for Forging 05 is a conservative estimate.

4. NICKEL CONTENT

The Pressurized Thermal Shock (PTS) Rule, 10CFR50.61,[7] requires that the nickel content be the best-estimate value. The nickel content for Weld 05B in the NA2 reactor vessel is not available, therefore, it is necessary to determine a best-estimate nickel content based on data from similar material.

According to the PTS Rule, the nickel content shall be obtained as follows:

1. Determine the mean value of measured values for welds fabricated using the same heat number as that of the critical reactor vessel weld.

2. If the above values are not available, the upper limiting values given in the material specifications to which the reactor vessel was built may be used.

3. If the above are not available, conservative estimates (mean plus one standard deviation) based on generic data (data from reactor vessels fabricated to the same material specification in the same shop and in the same time period) may be used if justification is provided.

4. If none of the above alternatives are available, 1.0 percent nickel must be assumed.

For Weld 05B in the NA2 reactor vessel, no data is available to satisfy the first two alternatives described above, however, nickel content data are available for Rotterdam welds that satisfy the requirements of Alternative 3. The nickel content for all welds fabricated in the same material specification in the same shop and in the same time period are listed in Table 4-1. The mean was found to be 0.10 weight percent with a standard deviation of 0.01. By the definition described in Alternative 3, the conservative estimate (mean plus one standard deviation) for the nickel content of NA2 Weld 05B is 0 11 percent.

Table 4-1. Nickel Content Data for Rotterdam Weld Metals Used to Determine Conservative Estimate Nickel Content

Plant	Weld Wire Type	Weld Wire Heat No.	Flux Type	Flux Lot No.	Ni, wt%	Reference
Ringhals Unit 2	S4 Mo	1725	SMIT 89	2275	0.084	WCAP-8216[8]
Sequoyah Unit 2	S4 Mo	4278	SMIT 89	1211	0.11	WCAP-8513[9]

$$\text{Mean} = 0.10$$

$$\sigma = 0.01$$

$$\text{Mean} + \sigma = 0.11$$

5. REVISED TABLES 1 AND 2

This section contains the revised Tables for Pressurized Thermal Shock and Upper Shelf Energy in the closure letter to Generic Letter 92-01, Revision 1, for NA1 and NA2.

Table 5-1. North Anna Unit 1 -- Data Summary for Pressurized Thermal Shock Calculation

Beltline Material	Heat No.	IS Neutron Fluence at 30.7 EFPY	IRT_{NDT} F	Method of Determin. IRT_{NDT}	Chemistry Factor	Method of Determin. CF	%Cu	%Ni
Nozzle Belt Shell Forging 05	990286/ 295213	2.77E+18[a]	+6	MTEB 5-2[b]	121.5	RG1.99 Table 2[e]	0.16[g]	0.74[g]
Interm. Shell Forging 04	990311/ 298244	3.95E+19[a]	+17	Plant Specific[c]	86	RG1.99 Table 2[e]	0.12[g]	0.82[g]
Lower Shell Forging 03	990400/ 292332	3.95E+19[a]	+38	Plant Specific[c]	88.9	Calculated[f]	0.15[g]	0.80[g]
Weld 05A NB to IS Circ. Weld (OD 94%)	25295	2.77E+18[a]	0	Generic[d]	143.8	RG1.99 Table 1[e]	0.30[g]	0.17[h]
Weld 05B NB to IS Circ. Weld (ID 6%)	4278	2.77E+18[a]	0	Generic[d]	59.5	RG1.99 Table 1[e]	0.11[g]	0.11[h]
Weld 04 IS to LS Circ. Weld	25531	3.95E+19[a]	+19	Plant Specific[c]	93.1	Calculated[f]	0.09[g]	0.11[g]

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

NOTES:

a. Values obtained from WCAP-11777.[10] (Nozzle belt shell forging and nozzle belt shell-to-intermediate shell circumferential weld fluences are 7% of maximum vessel inner surface fluence.)

b. Initial reference temperature was determined in accordance with MTEB 5-2 guidelines for cases where the reference temperature was not determined using ASME Boiler and Pressure Vessel Code, Section III, NB-2331,[11] methodology.

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

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

e. Chemistry factor was determined from the chemistry factor tables in Regulatory Guide 1.99, Revision 2.[12]

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

g. Chemical content obtained from BAW-1911, Revision 1.[13]

h. Chemical content obtained from Sequoyah Units 1 and 2 data.[9,14] (Same weld wire heat numbers.)

Table 5-2. North Anna Unit 1 -- Data Summary for Upper-Shelf Energy Calculation

Beltline Material	Heat No.	Material Type	1/4T USE at 30.7 EFPY	1/4T Neutron Fluence at 30.7 EFPY	Unirrad. USE	Method of Determin. Unirrad. USE
Nozzle Belt Shell Forging 05	990286/ 295213	A 508-2	62	1.75E+18[a]	74	See Section 3[b]
Interm. Shell Forging 04	990311/ 298244	A 508-2	68	2.49E+19[a]	92	Direct[c]
Lower Shell Forging 03	990400/ 292332	A 508-2	60	2.49E+19[a]	85	Direct[c]
Weld 05A NB to IS Circ. Weld (OD 94%)	25295	SMIT 89, SAW	78	1.75E+18[a]	111	Sister Plant[d]
Weld 05B NB to IS Circ. Weld (ID 6%)	4278	SMIT 89, SAW	---[e]	1.75E+18[a]	105	Sister Plant[d]
Weld 04 IS to LS Circ. Weld	25531	SMIT 89, SAW	73	2.49E+19[a]	102	Direct[c]

Table 5-2. (cont.) North Anna Unit 1 -- Data Summary for Upper-Shelf Energy Calculation

NOTES:

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

b. The unirradiated USE was determined on the basis of a comparison with similar materials to the beltline material. (See Section 3.)

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

d. The unirradiated USE was determined using reported data from other plants with the same weld wire heat number (Sequoyah Units 1 and 2[9,14]).

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

Table 5-3. North Anna Unit 2 -- Data Summary for Pressurized Thermal Shock Calculation

Beltline Material	Heat No.	IS Neutron Fluence at 32 EFPY	IRT_{NDT} F	Method of Determin. IRT_{NDT}	Chemistry Factor	Method of Determin. CF	%Cu	%Ni
Nozzle Belt Shell Forging 05	990598/ 291396	3.13E+18[a]	+9	MTEB 5-2[b]	51	RG1.99 Table 2[e]	0.08[g]	0.77[g]
Interm. Shell Forging 04	990496/ 292424	4.47E+19[a]	+75	Plant Specific[c]	47.9	Calculated[f]	0.09[g]	0.83[g]
Lower Shell Forging 03	990533/ 297355	4.47E+19[a]	+56	Plant Specific[c]	96	RG1.99 Table 2[e]	0.13[g]	0.83[g]
Weld 05A NB to IS Circ. Weld (OD 94%)	4278	3.13E+18[a]	0	Generic[d]	59.5	RG1.99 Table 1[e]	0.11[g]	0.11[h]
Weld 05B NB to IS Circ. Weld (ID 6%)	801	3.13E+18[a]	0	Generic[d]	87.8	RG1.99 Table 1[e]	0.18[g]	0.11[i]
Weld 04 IS to LS Circ. Weld	716126	4.47E+19[a]	-48	Plant Specific[c]	10.4	Calculated[f]	0.09[g]	0.08[g]

Table 5-3. (cont.) North Anna Unit 2 -- Data Summary for Pressurized Thermal Shock Calculation

NOTES:

a. Values obtained from WCAP-12497.[15] (Nozzle belt shell forging and nozzle belt shell-to-intermediate shell circumferential weld fluences are 7% of maximum vessel inner surface fluence.)

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

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

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

e. Chemistry factor was determined from the chemistry factor tables in Regulatory Guide 1.99, Revision 2.

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

g. Chemical content obtained from BAW-1911, Revision 1.

h. Chemical content obtained from Sequoyah Unit 2 data.[9] (Same weld wire heat number.)

i. Conservative estimate (mean plus one standard deviation) determined using data from other plants (Ringhals Unit 2 and Sequoyah Unit 2) with similar materials to the beltline material, i.e., data from reactor vessels fabricated to the same material specification in the same shop and in the same time period). (See Section 4.)

Table 5-4. North Anna Unit 2 -- Data Summary for Upper-Shelf Energy Calculation

Beltline Material	Heat No.	Material Type	1/4T USE at 32 EFPY	1/4T Neutron Fluence at 32 EFPY	Unirrad. USE	Method of Determin. Unirrad. USE
Nozzle Belt Shell Forging 05	990598/ 291396	A 508-2	64	1.98E+18[a]	74	Equiv. to Forging 04[b]
Interm. Shell Forging 04	990496/ 292424	A 508-2	56	2.82E+19[a]	74	Direct[c]
Lower Shell Forging 03	990533/ 297355	A 508-2	58	2.82E+19[a]	80	Direct[c]
Weld 05A NB to IS Circ. Weld (OD 94%)	4278	SMIT 89, SAW	87	1.98E+18[a]	105	Sister Plant[d]
Weld 05B NB to IS Circ. Weld (ID 6%)	801	SMIT 89, SAW	---[e]	1.98E+18[a]	---[e]	---[e]
Weld 04 IS to LS Circ. Weld	716126	LW 320, SAW	76	2.82E+19[a]	107	Direct[c]

Table 5-4. (cont.) North Anna Unit 2 -- Data Summary for Upper-Shelf Energy Calculation

NOTES:

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

b. Letter from W. L. Stewart, Virginia Electric and Power Company, to U. S. Nuclear Regulatory Commission, Subject: Virginia Electric and Power Company North Anna Power Station Unit 2 Selection of Limiting Forged Material for Low Upper-Shelf Energy Considerations, dated December 29, 1992.[5]

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

d. The unirradiated USE was determined using reported data from other plants with the same weld wire heat number (Sequoyah Unit 2[9]).

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

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

May 31, 1994

Docket Nos. 50-338
and 50-339

Mr. J.P. O'Hanlon
Senior Vice President - Nuclear
5000 Dominion Blvd.
Glen Allen, Virginia 23060

Dear Mr. O'Hanlon:

SUBJECT: GENERIC LETTER (GL) 92-01, REVISION 1, "REACTOR VESSEL STRUCTURAL INTEGRITY," VIRGINIA ELECTRIC AND POWER COMPANY (VEPCO) NORTH ANNA POWER STATION, UNITS NO. 1 AND NO. 2 (NA-1&2) (TAC NOS. M83488 AND M83489)

By letters dated May 29, 1992, September 28, 1992, October 22, 1992, December 29, 1992, September 23, 1993, and February 9, 1994, you provided a response to GL 92-01, Revision 1. The NRC staff has completed its review of your responses.

The GL is part of the staff's program to evaluate reactor vessel integrity for Pressurized Water Reactors (PWRs) and Boiling Water Reactors (BWRs). The information provided in response to GL 92-01, including previously docketed information, is being used to confirm that licensees satisfy the requirements and commitments necessary to ensure reactor vessel integrity for their facilities.

A substantial amount of information was provided in response to GL 92-01, Revision 1. These data have been entered into a computerized data base designated the Reactor Vessel Integrity Database (RVID). The RVID contains the following tables: a pressurized thermal shock (PTS) table for PWRs, a pressure-temperature limits table for BWRs and an upper-shelf energy (USE) table for PWRs and BWRs. Enclosure 1 provides the PTS tables. Enclosure 2 provides the USE tables for NA-1&2, and Enclosure 3 provides a key for the nomenclature used in the tables. The tables include the data necessary to perform USE and RT_{pts} evaluations. These data were taken from your responses to GL 92-01 and previously docketed information. References to the specific source of the data are provided in the tables.

We have determined that additional data is required to confirm that the USE at end-of-life (EOL) for one of your beltline materials, forging 05, for NA-1, is greater than 50 ft-lb because you have provided a generic mean value for the unirradiated USE. These types of values are unacceptable because they do not consider material variability. When the unirradiated USE for a particular material has not been determined, you can set the USE equal to the lower tolerance limit calculated for the group of similar materials. The unirradiated USE should be determined such that there exists 95% confidence

that at least 95% of the population is greater than the lower tolerance limit. If the lower tolerance limit results in a projected USE at EOL of less than 50 ft-lb, then you must demonstrate, in accordance with Appendix G, 10 CFR Part 50, that lower values of USE will provide margins of safety against fracture equivalent to those required by Appendix G of Section III of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code. We request that, within 30 days receipt of this letter, you submit a schedule for providing this required data.

Additionally, we have determined that additional data is required to confirm the value provided for the nickel content of weld 05B of the NA-2 reactor vessel. The value of 0.10 provided in the GL 92-01 submittal was cited as an "estimated" value. However, the supporting data and methodology for determining the estimated value were not provided. The Pressurized Thermal Shock (PTS) Rule, 10 CFR 50.61, requires that the amounts of copper and nickel be best-estimate values. According to the PTS Rule, a mean value is acceptable for welds fabricated using the same heat number as that which matches the critical reactor vessel weld. If these values are unavailable, upper limiting values given in the material specifications to which the reactor vessel was built may be used. If not available, conservative estimates (mean plus one standard deviation) based on generic data (data from reactor vessels fabricated to the same material specification in the same shop as your vessel and in the same time period) may be used if justification is provided. If none of these alternatives are available, 1.0 percent nickel must be assumed. We request that you provide the Westinghouse Owners Group (WOG) data that was used to determine the amount of nickel and that you determine the best-estimate amount of nickel in accordance with the PTS Rule, 10 CFR 50.61, within 30 days of receipt of this letter.

Further, we request that you verify that the information you have provided for NA-1&2 has been accurately entered in the summary file. If no comments are made in your response to this request, the staff will use the information in the tables for future NRC assessments of your reactor pressure vessels. Once your response is received and your schedule is determined to be satisfactory, the staff will consider your actions related to GL 92-01, Revision 1, to be complete. When your analyses are submitted, they will be reviewed as plant-specific licensing actions.

The information requested by this letter is within the scope of the overall burden estimated in GL 92-01, Revision 1, "Reactor Vessel Structural Integrity, 10 CFR 50.54(f)." The estimated average number of burden hours is 200 person hours for each addressee's response. This estimate pertains only to the identified response-related matters and does not include the time

required to implement actions required by the regulations. This action is covered by the Office of Management and Budget Clearance Number 3150-0011, which expires June 30, 1994.

Sincerely,

Leon B Engle

Leon B. Engle, Project Manager
Project Directorate II-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Pressurized Thermal Shock Tables
2. Upper-Shelf Energy Tables
3. Nomenclature Key

cc w/enclosures:
See next page

Mr. J. P. O'Hanlon
Virginia Electric & Power Company

North Anna Power Station
Units 1 and 2

cc:
Mr. William C. Porter, Jr.
County Administrator
Louisa County
P.O. Box 160
Louisa, Virginia 23093

Michael W. Maupin, Esq.
Hunton and Williams
Riverfront Plaza, East Tower
951 E. Byrd Street
Richmond, Virginia 23219

Dr. W. T. Lough
Virginia State Corporation Commission
Division of Energy Regulation
P.O. Box 1197
Richmond, Virginia 23209

Old Dominion Electric Cooperative
4201 Dominion Blvd.
Glen Allen, Virginia 23060

Mr. M. L. Bowling, Manager
Nuclear Licensing & Programs
Virginia Electric and Power Company
Innsbrook Technical Center
5000 Dominion Blvd.
Glen Allen, Virginia 23060

Office of the Attorney General
Supreme Court Building
101 North 8th Street
Richmond, Virginia 23219

Senior Resident Inspector
North Anna Power Station
U.S. Nuclear Regulatory Commission
Route 2, Box 78
Mineral, Virginia 231172

Robert B. Strobe, M.D., M.P.H.
State Health Commissioner
Office of the Commissioner
Virginia Department of Health
P.O. Box 2448
Richmond, Virginia 23218

Regional Administrator, RII
U.S. Nuclear Regulatory Commission
101 Marietta Street, N.W., Suite 2900
Atlanta, Georgia 30323

Mr. J. A. Stall, Manager
North Anna Power Station
P.O. Box 402
Mineral, Virginia 23117

Summary File for Pressurized Thermal Shock

Plant Name	Beltline Ident.	Heat No. Ident.	ID Neut. Fluence at EOL	IRT_{ndt}	Method of Determin. IRT_{ndt}	Chemistry Factor	Method of Determin. CF	%Cu	%Ni
North Anna 1	Nozzle shell forging 05	990286/ 295213	2.51E18	6°F	MTEB 5-2	121.5	Table	0.16	0.74
EOL: 4/1/2018	Int. shell forging 04	990311/ 298244	3.95E19	17°F	Plant Specific	86	Table	0.12	0.82
	Lower shell forging 03	990400/ 292332	3.95E19	38°F	Plant Specific	73.503	Calculated	0.16	0.80
	Weld 04	25531	3.95E19	19°F	Plant Specific	93.089	Calculated	0.09	0.11
	Weld 05A	25295	2.78E18	0°F	Generic	138.5	Table	0.30	0.17
	Weld 05B	4278	2.78E18	0°F	Generic	58.5	Table	0.11	0.11

References

The nickel contents for weld 05A and 05B are values from Sequoyah 1&2. (Same weld wire heat numbers).

Chemical composition and IRT_{ndt} data are from BAW-2168, which is attached to the GL 92-01 response.

Fluence data:

WCAP-11777: ID EOL fluence is 3.95E19 n/cm²

Table 2-1 of BAW-2146, which is attached to December 27, 1991, letter from W. L. Stewart (VPCo) to USNRC Document Control Desk, subject: Request to Change Technical Specifications: Pressure/Temperature Limitations, Low Temperature/Overpressure Protection System Setpoints, states that forging 05, and welds 05A and 05B have fluences that differ from forgings 03 and 04, and weld 04.

Note:

Chemical composition values for forging as are averages from the beltline material data and the surveillance data.

A margin of 69°F (σ_i = 20°F $\sigma\Delta$ = 28°F) has to be used for weld 05A and weld 05B for which a generic IRT_{ndt} of 0°F has been derived.

Summary File for Pressurized Thermal Shock

Plant Name	Beltline Ident.	Heat No. Ident.	ID Neut. Fluence at EOL	IRT_{ndt}	Method of Determin. IRT_{ndt}	Chemistry Factor	Method of Determin. CF	%Cu	%Ni
North Anna 2 EOL: 8/21/2020	Upper shell forging 05	990598/ 291396	4.47E19	9°F	MTEB 5-2	51	Table	0.08	0.77
	Int. shell forging 04	990496/ 292424	4.47E19	75°F	Plant Specific	35.112	Calculated	0.10	0.85
	Lower shell forging 03	990533/ 207355	4.47E19	56°F	Plant Specific	96	Table	0.13	0.83
	Weld 04	716126	4.47E19	-48°F	Plant Specific	10.398	Calculated	0.09	0.08
	Weld 05A	4278	4.47E19	0°F	Generic	58.5	Table	0.11	0.11
	Weld 05B	801	4.47E19	0°F	Generic	87.0	Table	0.18	0.10 [7]

References -

The nickel content for weld 05A is from Sequoyah 2 (the same weld wire heat number and the same flux).

BAW-2168, which is attached to June 29, 1992, letter from W. L. Stewart (VPCo) to USNRC Document Control Desk, subject: Response to Generic Letter 92-01, Reactor Vessel Structural Integrity, contains chemical composition and the initial RT_{ndt} (IRT_{ndt}) data for all the beltline materials

Fluence is from Table 6-13 of WCAP-12497

Note:

Chemical composition values for forging 04 are averages from the beltline material data and the surveillance data.

A margin of 69°F (σ_i = 20°F σ_Δ = 28°F) has to be used for weld 05A and weld 05B for which a generic IRT_{ndt} of 0°F has been derived.

[7] Additional information required to confirm value

Summary File for Upper Shelf Energy

Plant Name	Beltline Ident.	Heat No.	Material Type	1/4T USE at EOL	1/4T Neutron Fluence at EOL	Unirrad. USE	Method of Determin. Unirrad. USE
North Anna 1	Nozzle shell forging 05	990286/ 295213	A 508-2	62	1.74E18	75[7]	Generic
EOL: 4/1/2018	Int. shell forging 04	990311/ 298244	A 508-2	68	2.49E19	92	Direct
	Lower shell forging 03	990400/ 292332	A 508-2	58	2.49E19	85	Direct
	Circ. Weld Weld 04	25531	SMIT 89, SAW	71	2.49E18	102	Direct
-	Nozzle to Int. Shell Weld 05A	25295	SMIT 89, SAW	78	1.74E18	111	Sister Plant
	Nozzle to Int. Shell Weld 05B	4278	SMIT 89, SAW	---	1.74E18	105	Sister Plant

References

The fluence data for weld 04 is from June 29, 1992 letter to NRC (Response to GL 92-01); fluence data for other materials are from September 23, 1993 letter to NRC (Response to GL 92-01 RAI).

Chemical composition and UUSE data for forging 03 are from BAW-2168, which is attached to the GL 92-01 response.

UUSE data for forging 04 and weld 04 are from BAW-1911, Rev. 1.

Note: Weld 05A is 94% of thickness of the Nozzle to intermediate shell weld and 05B is the remainder. Therefore, it is not necessary to evaluate the EOL USE for weld 05B because it is not at the 1/4T location.

[7]Additional information required to confirm value

Summary File for Upper Shelf Energy

Plant Name	Beltline Ident.	Heat No.	Material Type	1/4T USE at EOL	1/4T Neutron Fluence at EOL	Unirrad. USE	Method of Determin. Unirrad. USE
North Anna 2 EOL: 8/21/2020	Upper shell forging 05	990598/ 291396	A 508-2	64	2.0E18	74	Equiv. to forging 04
	Int. shell forging 04	990496/ 292424	A 508-2	51	2.82E19	74	Direct
	Lower shell forging 03	990533/ 207355	A 508-2	58	2.82E19	80	Direct
	Weld 04	716126	LW 320, SAW	69	2.82E19	107	Direct
	Weld 05A	4278	SMIT 89, SAW	86	2.82E19	105	Sister Plant
	Weld 05B	801	SMIT 89, SAW	---	2.82E19	---	---

References

BAW-2168, which is attached to June 29, 1992, letter from W. L. Stewart (VPCo) to USNRC Document Control Desk, subject: Response to Generic Letter 92-01, Reactor Vessel Structural Integrity, contains chemical composition data for all the beltline materials. However, it contains UUSEs for forging 04 and weld 04 only.

Fluence and UUSEs for forgings 05 and 03 are from December 29, 1992 letter to NRC.

Note: Weld 05A is 94% of thickness of the Nozzle to intermediate shell weld and 05B is the remainder. Therefore, it is not necessary to evaluate the EOL USE for weld 05B because it is not at the 1/4T location.

PRESSURIZED THERMAL SHOCK AND USE TABLES FOR ALL PWR PLANTS

NOMENCLATURE

Pressurized Thermal Shock Table

Column 1: Plant name and date of expiration of license.
Column 2: Beltline material location identification.
Column 3: Beltline material heat number; for some welds that a single-wire or tandem-wire process has been reported, (S) indicates single wire was used in the SAW process, (T) indicates tandem wire was used in the SAW process.
Column 4: End-of-life (EOL) neutron fluence at vessel inner wall; cited directly from inner diameter (ID) value or calculated by using Regulatory Guide (RG) 1.99, Revision 2 neutron fluence attenuation methodology from the quarter thickness (T/4) value reported in the latest submittal (GL 92-01, PTS, or P/T limits submittals).
Column 5: Unirradiated reference temperature.
Column 6: Method of determining unirradiated reference temperature (IRT).

Plant-Specific
This indicates that the IRT was determined from tests on material removed from the same heat of the beltline material.

MTEB 5-2
This indicates that the unirradiated reference temperature was determined from following MTEB 5-2 guidelines for cases where the IRT was not determined using American Society of Mechanical Engineers Boiler and Pressure Vessel Code, Section III, NB-2331, methodology.

Generic
This indicates that the unirradiated reference temperature was determined from the mean value of tests on material of similar types.

Column 7: Chemistry factor for irradiated reference temperature evaluation.
Column 8: Method of determining chemistry factor

Table
This indicates that the chemistry factor was determined from the chemistry factor tables in RG 1.99, Revision 2.

Calculated
This indicates that the chemistry factor was determined from surveillance data via procedures described in RG 1.99, Revision 2.

Column 9: Copper content; cited directly from licensee value except when more than one value was reported. (Staff used the average value in the latter case.)

<u>No Data</u>
This indicates that no copper data has been reported and the default value in RG 1.99, Revision 2, will be used by the staff.

Column 10: Nickel content; cited directly from licensee value except when more than one value was reported. (Staff used the average value in the latter case.)

<u>No Data</u>

This indicates that no nickel data has been reported and the default value in RG 1.99, Revision 2, will be used by the staff.

Upper Shelf Energy Table

Column 1: Plant name and date of expiration of license.
Column 2: Beltline material location identification.
Column 3: Beltline material heat number; for some welds that a single-wire or tandem-wire process has been reported, (S) indicates single wire was used in the SAW process. (T) indicates tandem wire was used in the SAW process.
Column 4: Material type; plate types include A 533B-1, A 302B, A 302B Mod., and forging A 508-2; weld types include SAW welds using Linde 80, 0091, 124, 1092, ARCOS-B5 flux, Rotterdam welds using Graw Lo, SMIT 89, LW 320, and SAF 89 flux, and SMAW welds using no flux.
Column 5: EOL upper-shelf energy (USE) at T/4; calculated by using the EOL fluence and either the cooper value or the surveillance data. (Both methods are described in RG 1.99, Revision 2.)

<u>EMA</u>
This indicates that the USE issue may be covered by either owners group or plant-specific equivalent margins analyses.

Column 6: EOL neutron fluence at T/4 from vessel inner wall; cited directly from T/4 value or calculated by using RG 1.99, Revision 2 neutron fluence attenuation methodology from the ID value reported in the latest submittal (GL 92-01, PTS, or P/T limits submittals).

Column 7: Unirradiated USE.

<u>EMA</u>
This indicates that the USE issue may be covered by either owners group or plant-specific equivalent margins analyses.

Column 8: Method of determining unirradiated USE

<u>Direct</u>
For plates, this indicates that the unirradiated USE was from a transverse specimen. For welds, this indicates that the unirradiated USE was from test date.

<u>65%</u>
This indicates that the unirradiated USE was 65% of the USE from a longitudinal specimen.

<u>Generic</u>
This indicates that the unirradiated USE was reported by the licensee from other plants with similar materials to the beltline material.

<u>NRC generic</u>
This indicates that the unirradiated USE was derived by the staff from other plants with similar materials to the beltline material.

<u>10, 30, 40, or 50 °F</u>
This indicates that the unirradiated USE was derived from Charpy test conducted at 10, 30, 40, or 50 °F.

<u>Surv. Weld</u>
This indicates that the unirradiated USE was from the surveillance weld having the same weld wire heat number.

<u>Equiv. to Surv. Weld</u>
This indicates that the unirradiated USE was from the surveillance weld having different weld wire heat number.

<u>Sister Plant</u>
This indicates that the unirradiated USE was derived by using the reported value from other plants with the same weld wire heat number.

<u>Blank</u>
indicates that there is insufficient data to determine the unirradiated USE.

6. REFERENCES

1. VEPCO North Anna Power Station Units 1 and 2, Final Safety Analysis Report, USNRC Docket Nos. 50-338 and 50-339.

2. J. C. Schmertz, "Analysis of Capsule U from the Virginia Power Company North Anna Unit 1 Reactor Vessel Radiation Surveillance Program, North Anna Unit 1 Reactor Vessel Heatup and Cooldown Limit Curves for Normal Operation," WCAP-11791, Westinghouse Electric Corporation, Pittsburgh, Pennsylvania, May 1988.

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

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

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

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

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

8. S. E. Yanichko and D. J. Lege, "Swedish State Power Board Ringhals Unit No. 2 Reactor Vessel Radiation Surveillance Program," WCAP-8216, Westinghouse Electric Corporation, Pittsburgh, Pennsylvania, November 1973.

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

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

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

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

13. A. L. Lowe, Jr., "Reactor Pressure Vessel and Surveillance Program Materials Licensing Information for North Anna Units 1 and 2," BAW-1991, Revision 1, Babcock & Wilcox Nuclear Power Division, Lynchburg, Virginia, August 1986.

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

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

7. CERTIFICATION

This report accurately responds to the request for information stated in the closure letter to Generic Letter 92-01, Revision 1, for North Anna Unit 1 and Unit 2.

M J DeVan 7/27/94

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

This report was reviewed and found to be accurate.

L B Gross 7/27/94

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

Verification of independent review.

K E Moore 7-28-94

K. E. Moore, Manager Date
Materials and Structural Analysis Unit

This report is approved for release.

JKelly for 7/28/94

D. L. Howell, Program Manager Date
Owners Group Projects