BAW-2168

June 1992

RESPONSE TO GENERIC LETTER 92-01
FOR VIRGINIA ELECTRIC & POWER COMPANY
NORTH ANNA UNIT 1 AND
NORTH ANNA UNIT 2

by

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BWNS Document No. 77-2168-00
(See Section 6 for document signatures.)

Prepared for

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CONTENTS

1. INTRODUCTION

This report provides a response to the Nuclear Regulatory Commission (NRC) Generic Letter 92-01 for Virginia Electric & Power Company; North Anna Unit 1 and North Anna Unit 2.

Generic Letter 92-01, Revision 1, shown in Section 2 of this report, was issued by the NRC on March 6, 1992 and addressed to all holders of nuclear power plant operating licenses. The generic letter was issued to obtain information from the licensees to enable the NRC to assess the degree of compliance with regulatory requirements regarding reactor vessel integrity. Response is required within 120 days of the issue date; this comes to July 4, 1992. This document provides the required information, insofar as it is available.

2. GENERIC LETTER

Generic Letter 92-01, Revision 1, is shown below.

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

March 6, 1992

TO: ALL HOLDERS OF OPERATING LICENSES OR CONSTRUCTION PERMITS FOR NUCLEAR POWER PLANTS (EXCEPT YANKEE ATOMIC ELECTRIC COMPANY, LICENSEE FOR THE YANKEE NUCLEAR POWER STATION)

SUBJECT: REACTOR VESSEL STRUCTURAL INTEGRITY, 10 CFR 50.54(f) (GENERIC LETTER 92-01, REVISION 1)

This letter replaces Generic Letter 92-01 dated February 28, 1992. The background information concerning NRC's assessment of embrittlement in the Yankee Nuclear Power Station reactor vessel was drafted by staff some months ago and has now been clarified and updated to better reflect the licensee's extensive technical efforts regarding reactor vessel integrity. The section pertaining to required information has not changed.

The U.S. Nuclear Regulatory Commission (NRC) is issuing this generic letter to obtain information needed to assess compliance with requirements and commitments regarding reactor vessel integrity in view of certain concerns raised in the staff's review of reactor vessel integrity for the Yankee Nuclear Power Station. In Section 50.60(a) of Title 10 of the Code of Federal Regulations (10 CFR 50.60(a)), the NRC requires that licensees for all light water nuclear power reactors meet fracture toughness requirements and have a material surveillance program for the reactor coolant pressure boundary. These requirements are set forth in Appendices G and H to 10 CFR Part 50. In 10 CFR 50.60(b), where the requirements of Appendices G and H to 10 CFR Part 50 cannot be met, an exemption is necessary pursuant to 10 CFR 50.12. In 10 CFR 50.61 the NRC also provided fracture toughness requirements for protecting pressurized water reactors against pressurized thermal shock events. Licensees and permit holders have also made commitments in response to Generic Letter (GL) 88-11, "NRC Position on Radiation Embrittlement of Reactor Vessel Materials and its Impact on Plant Operations," to use the methodology in Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials," to predict the effects of neutron irradiation as required by Paragraph V.A of 10 CFR Part 50, Appendix G. The 10 CFR 50.60 and 10 CFR 50.61 requirements and GL 88-11 are in the overall regulatory program to maintain the structural integrity of the reactor vessel.

This generic letter is part of a program to evaluate reactor vessel integrity and take regulatory actions, if needed, to ensure that licensees and permit holders are complying with 10 CFR 50.60 and 10 CFR 50.61, and are fulfilling commitments made in response to GL 88-11. Enclosure 1 is a discussion of the applicable regulatory requirements. The NRC is requiring information on compliance under the provisions of 10 CFR 50.54(f).

Assessment of Embrittlement for the Yankee Nuclear Power Station Reactor Vessel

In an effort to resolve concerns regarding the neutron embrittlement of the Yankee reactor vessel, the staff performed a safety assessment of the Yankee reactor vessel. The staff found that the licensee for the Yankee Nuclear Power Station might not be in compliance with 10 CFR 50.60 and 10 CFR 50.61.

The staff found that the Charpy upper shelf energy of the Yankee reactor vessel material could be as low as 35.5 foot-pounds which is less than the 50 foot-pound value required in Appendix G to 10 CFR Part 50. However, the licensee for the Yankee Nuclear Power Station had not performed the actions required in Paragraphs IV.A.1 or V.C of Appendix G to 10 CFR Part 50. Since then, the licensee has performed an analysis in accordance with Paragraph IV.A.1 of Appendix G to 10 CFR Part 50 using criteria being developed by the American Society of Mechanical Engineers (ASME) to demonstrate margins of safety equivalent to those in the ASME Code.

The NRC expressed a concern regarding compliance with the requirements of Appendix H to 10 CFR Part 50. Section E 185 of the American Society for Testing and Materials (ASTM) Code requires that the licensee take sample specimens from actual material used in fabricating the beltline of the reactor vessel. These surveillance materials shall include one heat of base metal, one butt weld, and one weld "heat affected zone." The licensee for the Yankee Nuclear Power Station terminated the material surveillance program in 1965. Therefore, the Yankee Nuclear Power Station had no material surveillance program on July 26, 1983, when Appendix H to 10 CFR Part 50 became effective. Further, the samples irradiated at Yankee Rowe before 1965 were comprised only of base metal.

The licensee for the Yankee Nuclear Power Station had used the methodology in draft Regulatory Guide 1.99, Revision 2, to predict the effects of neutron embrittlement. The staff raised concerns regarding the licensee's application of the methodology. The specific issues were (1) the irradiation temperature, (2) the chemistry composition of reactor vessel material, and (3) the results of the material surveillance program.

The irradiation temperature at the Yankee Nuclear Power Station is between 454 °F and 520 °F, which is below the nominal irradiation temperature of 550 °F used in developing Regulatory Guide 1.99, Revision 2. A lower irradiation temperature increases the effect of neutron embrittlement. The regulatory guide indicates that for irradiation temperatures less than 525 °F, embrittlement effects should be considered to be greater than predicted by the methods of the guide. Adjustments that were made by the licensee were insufficient to account for this effect.

The results of the surveillance program from the Yankee Nuclear Power Station indicated that the increase in the reference temperature exceeds the mean-plus-two standard deviations as predicted by the procedures in Regulatory Guide 1.99, Revision 2. The regulatory guide states that the licensee should use credible surveillance data to predict the increase in reference temperature resulting from neutron irradiation.

The staff implemented RG 1.99, Revision 2, by issuing GL 88-11. In committing to GL 88-11, licensees have committed to calculate radiation embrittlement in accordance with the procedures documented in RG 1.99, Revision 2. To meet the limitations in Section 1.3 of the regulatory guide, the licensee should consider the effects on irradiation embrittlement during core critical operation with irradiation temperatures less than 525 °F. Section 2 of the regulatory guide states that the licensees should consider the effects of the results from its surveillance capsules.

The Summer 1972 Addenda of the 1971 Edition of Section III of the ASME Boiler and Pressure Vessel Code are the earliest code requirements for testing materials to determine their unirradiated reference temperature. The Yankee reactor vessel was constructed in 1959 to ASME Code, Section VIII. Therefore, the unirradiated reference temperature could not be established in accordance with the requirements of the Summer 1972 Addenda. The licensee for the Yankee Nuclear Power Station extrapolated the available test results to determine an unirradiated reference temperature. The staff determined that the licensee's extrapolation was not conservative.

The chemical composition of the Yankee reactor vessel welds is unknown. The material's sensitivity to neutron embrittlement depends on its chemical content. The licensee assumed that the chemistry of its welds was equivalent to that of the BR-3 reactor vessel in Mol, Belgium. The heat number of the wire used to fabricate the Yankee welds was not available. The licensee was assuming a chemical composition that was not based on its plant-specific information, since the chemical composition, in particular, the amount of copper, depends upon the heat number of the weld wire.

These factors prompted the staff to find that the licensee for the Yankee Nuclear Power Station had not fully considered plant-specific information in assessing compliance with 10 CFR 50.61. When plant-specific information is considered, the Yankee reactor vessel may have exceeded the screening criteria in 10 CFR 50.61.

Upon conducting the Yankee Nuclear Power Station review, the staff became concerned about other licensee's compliance with 10 CFR 50.60 and 10 CFR 50.61 and fulfillment of commitments made in response to GL 88-11. Thus, the staff is issuing this generic letter to obtain information to assess compliance with these regulations and fulfillment of commitments. The staff is continuing to pursue this concern with the Yankee Atomic Electric Company. Therefore, the Yankee Atomic Electric Company need not respond to this generic letter.

Required Information

Portions of the following information requested are not applicable to all addressees. The responses provided should, in these cases, indicate that the requested information is not applicable and why it is not applicable.

1. Certain addressees are requested to provide the following information regarding Appendix H to CFR Part 50:

   Addressees who do not have a surveillance program meeting ASTM E 185-73, -79, or -82 and who do not have an integrated surveillance program approved by the NRC (see Enclosure 2), are requested to describe actions taken or to be taken to ensure compliance with Appendix H to 10 CFR Part 50. Addressees who plan to revise the surveillance program to meet Appendix H to 10 CFR Part 50 are requested to indicate when the revised program will be submitted to the NRC staff for review. If the surveillance program is not to be revised to meet Appendix H to 10 CFR Part 50, addressees are requested to indicate when they plan to request an exemption from Appendix H to 10 CFR Part 50 under 10 CFR 50.60(b).

2. Certain addressees are requested to provide the following information regarding Appendix G to 10 CFR Part 50:

   a. Addressees of plants for which the Charpy upper shelf energy is predicted to be less than 50 foot-pounds at the end of their licenses using the guidance in Paragraphs C.1.2 or C.2.2 in Regulatory Guide 1.99, Revision 2, are requested to provide to the NRC the Charpy upper shelf energy predicted for December 16, 1991, and for the end of their current license for the limiting beltline weld and the plate or forging and are requested to describe the actions taken pursuant to Paragraphs IV.A.1 or V.C of Appendix G to 10 CFR Part 50.

   b. Addressees whose reactor vessels were constructed to an ASME Code earlier than the Summer 1972 Addenda of the 1971 Edition are requested to describe the consideration given to the following material properties in their evaluations performed pursuant to 10 CFR 50.61 and Paragraph III.A of 10 CFR Part 50, Appendix G:

      (1) the results from all Charpy and drop weight tests for all unirradiated beltline materials, the unirradiated reference temperature for each beltline material, and the method of determining the unirradiated reference temperature from the Charpy and drop weight test;

      (2) the heat treatment received by all beltline and surveillance materials;

      (3) the heat number for each beltline plate or forging and the heat number of wire and flux lot number used to fabricate each beltline weld;

(4) the heat number for each surveillance plate or forging and the heat number of wire and flux lot number used to fabricate the surveillance weld;

(5) the chemical composition, in particular the weight in percent of copper, nickel, phosphorous, and sulfur for each beltline and surveillance material; and

(6) the heat number of the wire used for determining the weld metal chemical composition if different than Item (3) above.

3. Addressees are requested to provide the following information regarding commitments made to respond to GL 88-11:

   a. How the embrittlement effects of operating at an irradiation temperature (cold leg or recirculation suction temperature) below 525 °F were considered. In particular licensees are requested to describe consideration given to determining the effect of lower irradiation temperature on the reference temperature and on the Charpy upper shelf energy.

   b. How their surveillance results on the predicted amount of embrittlement were considered.

   c. If a measured increase in reference temperature exceeds the mean-plus-two standard deviations predicted by Regulatory Guide 1.99, Revision 2, or if a measured decrease in Charpy upper shelf energy exceeds the value predicted using the guidance in Paragraph C.1.2 in Regulatory Guide 1.99, Revision 2, the licensee is requested to report the information and describe the effect of the surveillance results on the adjusted reference temperature and Charpy upper shelf energy for each beltline material as predicted for December 16, 1991, and for the end of its current license.

Reporting Requirements

Pursuant to Section 182a of the Atomic Energy Act of 1954, as amended, and 10 CFR 50.54(f), each addressee shall submit a letter within 120 days of the date of this generic letter providing the information described under "Required Information." The letter shall be addressed to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555, under oath or affirmation. A copy shall also be submitted to the appropriate Regional Administrator. This generic letter requests information that will enable the NRC to verify that the licensee is complying with its current licensing basis regarding reactor vessel fracture toughness and material surveillance for the reactor coolant pressure boundary. Accordingly, an evaluation justifying this information request is not necessary under 10 CFR 50.54(f).

Backfit Discussion

This generic letter requests information that will enable the NRC staff to determine whether licensees are complying with their prior commitments and any license conditions regarding 10 CFR 50.60, 10 CFR 50.61, and GL 88-11. The staff is not establishing a new position for such compliance in this generic letter. The staff is requesting information to verify that the licensee is complying with its previously established commitments and is not establishing any new position. Therefore, this generic letter does not constitute a backfit and no documented evaluation or backfit analysis need be prepared.

Request for Voluntary Submittal of Impact Data

This request is covered by Office of Management and Budget Clearance Number 3150-0011, which expires May 31, 1994. The estimated average number of burden hours is 200 person hours for each addressee's response, including the time required to assess the requirements, search data sources, gather and analyze the data, and prepare the required letters. This estimated average number of burden hours pertains only to the identified response-related matters and does not include the time to implement the actions required by the regulations. Comments on the accuracy of this estimate and suggestions to reduce the burden may be directed to Ronald Minsk, Office of Information and Regulatory Affairs (3150-0011), NEOB-3019, Office of Management and Budget, Washington, DC 20503, and to the U.S. Nuclear Regulatory Commission, Information and Records Management Branch, Division of Information Support Services, Office of Information and Resources Management, Washington, DC 20555.

Although no specific request or requirement is intended, the following information would assist the NRC in evaluating the cost of complying with this generic letter:

(1) the licensee staff's time and costs to perform requested inspections, corrective actions, and associated testing;

(2) the licensee staff's time and costs to prepare the requested reports and documentation;

(3) the additional short-term costs incurred to address the inspection findings such as the costs of the corrective actions or the costs of down time; and

(4) an estimate of the additional long-term costs that will be incurred as a result of implementing commitments such as the estimated costs of conducting future inspections or increased maintenance.

If you have any questions about this matter, please contact one of the NRC technical contacts or the lead project manager listed below.

Sincerely,

James G. Partlow
Associate Director for Projects
Office of Nuclear Reactor Regulation

Enclosures:
1. Applicable Regulatory Requirements
2. Plants with Integrated Programs
3. List of Recently Issued Generic Letters

Technical Contacts:
Barry J. Elliot, NRR
(301) 504-2709

Keith R. Wichman, NRR
(301) 504-2757

Lead Project Manager:
Daniel G. McDonald, NRR
(301) 504-1408

Regulatory Requirements Applicable to

Reactor Vessel Structural Integrity

10 CFR 50.60

Pursuant to 10 CFR 50.60, all light water nuclear power reactors must meet the fracture toughness and material surveillance program requirements for the reactor coolant pressure boundary set forth in Appendices G and H to 10 CFR Part 50.

The fracture toughness of the reactor coolant pressure boundary required by 10 CFR 50.60 is necessary to provide adequate margins of safety during any condition of normal plant operation, including anticipated operational occurrences and system hydrostatic tests. The material surveillance program required by 10 CFR 50.60 monitors changes in the fracture toughness properties of ferritic materials in the reactor vessel beltline region of light water nuclear power reactors resulting from exposure of these materials to neutron irradiation and the thermal environment. Under the program, fracture toughness test data are obtained from material specimens exposed in surveillance capsules, which are withdrawn periodically from the reactor vessel.

Appendix G to 10 CFR Part 50 requires that the reactor vessel beltline materials must have Charpy upper shelf energy of no less than 50 ft-lb throughout the life of the vessel. Otherwise, licensees are required to provide demonstration of equivalent margins of safety in accordance with Paragraph IV.A.1 of Appendix G to 10 CFR Part 50 or perform actions in accordance with Paragraph V.C of Appendix G to 10 CFR Part 50.

Appendix H to 10 CFR Part 50 requires the surveillance program to meet the American Society for Testing and Materials (ASTM) Standard E 185, "Standard Practice for Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels." Further, Appendix H to 10 CFR Part 50 specifies the applicable edition of ASTM E 185. Appendix H to 10 CFR Part 50, as amended on July 26, 1983, requires that the part of the surveillance program conducted before the first capsule is withdrawn must meet the requirements of the 1973, the 1979, or the 1982 edition of ASTM E 185 that is current on the issue date of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code under which the reactor vessel was purchased. The licensee may also use later editions of ASTM E 185 which have been endorsed by the NRC. The test procedures and reporting requirements for each capsule withdrawal after July 26, 1983 must meet the requirements of the 1982 edition of ASTM E 185 to the extent practical for the configuration of the specimens in the capsule. The licensee may use either the 1973, the 1979, or the 1982 edition of ASTM E 185 for each capsule withdrawal before July 26, 1983.

Licensees, especially those with reactor vessels purchased before ASTM issued the 1973 edition of ASTM E 185, may have surveillance programs that do not meet the requirements of Appendix H to 10 CFR Part 50 but may have alternative surveillance programs. The licensee may use these alternative surveillance programs in accordance with 10 CFR 50.60(b) if the licensee has been granted an exemption by the Commission under 10 CFR 50.12.

The licensee must monitor the test results from the material surveillance program. According to Paragraph III.C of Appendix H to 10 CFR Part 50, the results of the surveillance program may indicate that a technical specifications change is required, either in the pressure-temperature limits or in the operating procedures required to meet the limits.

10 CFR 50.61

Pursuant to 10 CFR 50.61, there are fracture toughness requirements for protection against pressurized thermal shock events for pressurized water reactors. Licensees are required to perform an assessment of the projected values of reference temperature. If the projected reference temperature exceeds the screening criteria established in 10 CFR 50.61, licensees are required to submit an analysis and schedule for such flux reduction programs as are reasonably practicable to avoid exceeding the screening criteria. If no reasonably practicable flux reduction program will avoid exceeding the screening criteria, licensees shall submit a safety analysis to determine what actions are necessary to prevent potential failure of the reactor vessel if continued operation beyond the screening criteria is allowed. In 10 CFR 50.61(b)(1), as amended effective June 14, 1991 (56 Fed Reg 22300 et. seq., May 15, 1991), licensees are required to submit their assessment by December 16, 1991, if the projected reference temperature will exceed the screening criteria before the expiration of the operating license.

Plant-specific information is required to be considered in assessing the level of neutron embrittlement as specified in 10 CFR 50.61(b)(3). This information includes but is not limited to the reactor vessel operating temperature and surveillance results.

Prediction of Irradiation Embrittlement

Paragraph V.A of Appendix G to 10 CFR Part 50 requires the prediction of the effects of neutron irradiation on reactor vessel materials. The extent of neutron embrittlement depends on the material properties, thermal environment, and results of the material surveillance program. In Generic Letter 88-11, "NRC Position on Radiation Embrittlement of Reactor Vessel Materials and its Impact on Plant Operations," the staff stated that it will use the guidance in Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials," in estimating the embrittlement of the materials in the reactor vessel beltline. All licensees and permittees have responded to Generic Letter 88-11 committing to use the methodology in Regulatory Guide 1.99, Revision 2, in predicting the effects of neutron irradiation as required by Paragraph V.A of 10 CFR Part 50, Appendix G. The methodology in Regulatory Guide 1.99, Revision 2, is also the basis in 10 CFR 50.61 in projecting the reference temperature.

Enclosure 2

Plants With Integrated Surveillance Programs Approved By The NRC

Oconee Units 1, 2, and 3
Arkansas Nuclear One Unit 1
Rancho Seco
Three Mile Island Unit 1
Davis-Besse
Ginna
Point Beach Units 1 and 2
Surry Units 1 and 2
Turkey Point Units 3 and 4
Zion Units 1 and 2

3. METHOD OF RESPONSE

3.1. Organization

The Generic Letter presents the information requests in three sections (1, 2, and 3) further divided into a number of items. Ten distinct sections/items were identified, each of which are presented in a table. The tables are identified as follows:

Table	GL 92-01 Reference	Subject
(1)	Section 1	10CFR50, Appendix H; Adherence to RVSP Requirements
(2)	Section 2, Item a	10CFR50, Appendix G; C_VUSE Requirements
(3)	Section 2, Item b, ¶ (1)	10CFR50.61 and 10CFR50, Appendix G, III.A; Material Properties Related to PTS and Fracture Toughness Requirements [unirradiated Charpy and RT_{NDT} values]
(4)	Section 2, Item b, ¶ (2)	10CFR50.61 and 10CFR50, Appendix G, III.A; Material Properties Related to PTS and Fracture Toughness Requirements [material heat treatment]
(5)	Section 2, Item b, ¶ (3)	10CFR50.61 and 10CFR50, Appendix G, III.A; Material Properties Related to PTS and Fracture Toughness Requirements [beltline material identification]
(6)	Section 2, Item b, ¶ (4)	10CFR50.61 and 10CFR50, Appendix G, III.A; Material Properties Related to PTS and Fracture Toughness Requirements [surveillance material identification]
(7)	Section 2, Item b, ¶ (5)	10CFR50.61 and 10CFR50, Appendix G, III.A; Material Properties Related to PTS and Fracture Toughness Requirements [chemical composition]

Table	GL 92-01 Reference	Subject
(8)	Section 3, Item a	Generic Letter 88-11 Response Commitments; Effect of Irradiation Temperature
(9)	Section 3, Item b	Generic Letter 88-11 Response Commitments; Utilization of Surveillance Results
(10)	Section 3, Item c	Generic Letter 88-11 Response Commitments; Difference Between Measured and Predicted (Regulatory Guide 1.99, Revision 2) Embrittlement Effects

Each of the above ten tables were prepared for each of the four plants covered by this report. These tables are presented in Section 4 of this report.

3.2. Response Details

3.2.1 Abbreviations used in the response are as follows:

ART_{NDT}	Adjusted reference temperature
C_VUSE	Charpy upper-shelf energy
EOL	End of life
EST	Estimated value
NA	Not applicable
ND	Not determined
PTS	Pressurized thermal shock
RVSP	Reactor vessel surveillance program
RT_{NDT}	Reference temperature
ΔRT_{NDT}	Reference temperature shift
σ	Standard deviation

3.2.2 Material properties were determined at the ¼-thickness location, in accordance with 10CFR50, Appendix G, ¶ V.B, Footnote 2. Effects of neutron embrittlement were determined in accordance with the methods of

Regulatory Guide 1.99, Revision 2. The drop in C_vUSE was determined in accordance with Position 1. The End-of-Life (EOL) is taken as the time when 32 EFPY is achieved.

4. RESPONSE TO GENERIC LETTER 92-01

The following tables are submitted in response to the information requested in Generic Letter 92-01.

North Anna Unit 1

Table 1. Adherence to RVSP Requirements
Table 2. C_VUSE Requirements
Table 3. Unirradiated Charpy and RT_{NDT} Values
Table 4. Material Heat Treatment
Table 5. Beltline Material Identification
Table 6. Surveillance Material Identification
Table 7. Chemical Composition
Table 8. Effect of Irradiation Temperature
Table 9. Utilization of Surveillance Results
Table 10. Difference Between Measured and Predicted Embrittlement Effects

North Anna Unit 2

Table 1. Adherence to RVSP Requirements
Table 2. C_VUSE Requirements
Table 3. Unirradiated Charpy and RT_{NDT} Values
Table 4. Material Heat Treatment
Table 5. Beltline Material Identification
Table 6. Surveillance Material Identification
Table 7. Chemical Composition
Table 8. Effect of Irradiation Temperature
Table 9. Utilization of Surveillance Results
Table 10. Difference Between Measured and Predicted Embrittlement Effects

TABLE 3. GENERIC LETTER 92-01 RESPONSE: SECTION 2, ITEM b, ¶ (1)									
Subject: 10CFR50.61 and 10CFR50, Appendix G, III.A; Material Properties Related to PTS and Fracture Toughness Requirements									
Plant: North Anna Unit 1									
Column 1	Column 2				Column 3	Column 4	Column 5	C.6
Beltline Materials	Unirradiated Charpy Test Results				Unirrad. Dropwt. Test Results T_{NDT} F	Unirrad. RT_{NDT} F	Method of Determng RT_{NDT}	Notes
	Col. 2a	Col. 2b	Col. 2c	Col. 2d				
	C_v 10 F ft-lb	C_v 30 ft-lb F	C_v 50 ft-lb F	C_v 35 MLE F				
FORGING								
05	ND	ND	ND	ND	2	6	Estimated	(1)
04	ND	ND	ND	ND	-31	17	NB-2331	(1)
03	ND	ND	ND	ND	-13	38	NB-2331	(1)
WELD								
W05A	ND	ND	ND	ND	0	0	Estimated	(1)
W05B	ND	ND	ND	ND	0	0	Estimated	(1)
W04	ND	ND	ND	ND	-13	19	NB-2331	(1)

NOTES:

(1) BAW-1911, Revision 1

TABLE 4. GENERIC LETTER 92-01 RESPONSE: SECTION 2, ITEM b, ¶ (2)		
Subject: 10CFR50.61 and 10CFR50, Appendix G, III.A; Material Properties Related to PTS and Fracture Toughness Requirements -- APPLICABLE ONLY TO REACTOR VESSELS CONSTRUCTED TO AN ASME CODE EARLIER THAN THE 1971 EDITION, SUMMER 1972 ADDENDA		
Plant: North Anna Unit 1		
Column 1	Column 2	Col. 3
Material	Heat Treatment	Notes
BELTLINE MATERIALS		
Forging 05	1616-1725F-2½hr/WQ; 1202-1292F-7½hr/FC; 1130±25F-14¾hr/FC	(1,3)
Forging 04	1616-1725F-2½hr/WQ; 1202-1292F-7½hr/FC; 1130±25F-14¾hr/FC	
Forging 03	1616-1725F-2½hr/WQ; 1202-1292F-7½hr/FC; 1130±25F-14¾hr/FC	
Weld 05A	1130±25F-10¾hr/FC	
Weld 05B	1130±25F-10¾hr/FC	
Weld 04	1130±25F-10¾hr/FC	
SURVEILLANCE MATERIALS		
Forging 03	1616-1725F-2½hr/WQ; 1202-1292F-7½hr/FC; 1130±25F-14¾hr/FC	(2,3)
Weld W04	1130±25F-10¾hr/FC	

NOTES:

(1) Estimated based on review of surveillance data.
(2) BAW-1911, Revision 1
(3) WQ - water quench
FC - furnace cool

TABLE 1. GENERIC LETTER 92-01 RESPONSE: SECTION 1

Subject: 10CFR50, Appendix H; Adherence to RVSP Requirements

Plant: North Anna Unit 1

Question I:	Does RVSP meet ASTM E 185-73, E 185-79, or E 185-82?	Yes ✓ No □
Question II:	Is plant one of the following? ANO-1, Crystal River 3, Davis Besse, R. E. Ginna, Oconee-1, Oconee-2, Oconee-3, Point Beach-1, Point Beach-2, Rancho Seco, Surry-1, Surry-2, Turkey Point-3, Turkey Point-4, Zion-1, Zion-2.	Yes □ No ✓

IF ANSWER IS "YES" TO EITHER QUESTION I OR QUESTION II, PROCEED TO TABLE 2.

IF ANSWER IS "NO" TO BOTH QUESTION I AND QUESTION II, PROCEED TO QUESTION III AND QUESTION IV.

Question III: If plan is to revise RVSP to meet requirements of 10CFR50, Appendix H, when will revised RVSP be submitted to NRC?

Response:

Not Applicable (See Question I and II above)

Question IV: If plan is not to revise RVSP to meet requirements of 10CFR50, Appendix H, when will exemption from 10CFR50, Appendix H be requested from NRC?

Response:

Not Applicable (See Question I and II above)

NOTES: (1) WCAP-8771, RVSP per ASTM E185-73.

TABLE 2. GENERIC LETTER 92-01 RESPONSE: SECTION 2, ITEM a					
Subject: 10CFR50, Appendix G, C_vUSE Requirements					
Plant: North Anna Unit 1					
Column 1	Column 2		Column 3		Column 4
Limiting Material	Initial USE ft-lb	EFPY to reach C_vUSE<50 ft-lb	If Column 2 is within license period: C_vUSE at indicated time		Action taken per IV.A.1
			Column 3A	Column 3B	
			12/16/91	EOL	
LIMITING BELTLINE WELD					
Weld 05A	90(2)	>32	NA	NA	NA
LIMITING BELTLINE FORGING					
Forging 03	85	>32	NA	NA	NA

NOTES:

(1) Fluence values for C_vUSE taken at ¼-thickness.

(2) Estimated value.

NORTH ANNA UNIT 1

TABLE 5. GENERIC LETTER 92-01 RESPONSE: SECTION 2, ITEM b, ¶ (3)					
Subject: 10CFR50.61 and 10CFR50, Appendix G, III.A; Material Properties Related to PTS and Fracture Toughness Requirements -- APPLICABLE ONLY TO REACTOR VESSELS CONSTRUCTED TO AN ASME CODE EARLIER THAN THE 1971 EDITION, SUMMER 1972 ADDENDA					
Plant: North Anna Unit 1					
Column 1	Column 2	Column 3	Column 4	Column 5	C. 6
Beltline Plate or Forging	Heat Number	Beltline Weld	Weld Wire Heat	Weld Flux Lot	Notes
Forging 05 Forging 04 Forging 03	990286/295213 990311/298244 990400/292332	Weld 05A Weld 05B Weld 04	25295 (2) 4278 (4) 25531 (2)	1170 (3) 1211 (3) 1211 (3)	(1)

NOTES:

(1) BAW-1911, Revision 1

(2) SMIT 40

(3) SMIT 89

(4) S4 Mo

TABLE 6. GENERIC LETTER 92-01 RESPONSE: SECTION 2, ITEM b, ¶ (4)				
Subject: 10CFR50.61 and 10CFR50, Appendix G, III.A; Material Properties Related to PTS and Fracture Toughness Requirements -- APPLICABLE ONLY TO REACTOR VESSELS CONSTRUCTED TO AN ASME CODE EARLIER THAN THE 1971 EDITION, SUMMER 1972 ADDENDA				
Plant: North Anna Unit 1				
Column 1	Column 2	Column 3	Column 4	Column 5
Surveillance Plate or Forging Heat Number	Surveillance Weld	Weld Wire Heat	Weld Flux Lot	Notes
Forging 03 990400/292332	Weld 04	25531 (SMIT 40)	1211 (SMIT 89)	(1)

NOTES:

(1) BAW-1911, Revision 1

TABLE 7. GENERIC LETTER 92-01 RESPONSE: SECTION 2, ITEM b, ¶ (5)

Subject: 10CFR50.61 and 10CFR50, Appendix G, III.A; Material Properties Related to PTS and Fracture Toughness Requirements -- APPLICABLE ONLY TO REACTOR VESSELS CONSTRUCTED TO AN ASME CODE EARLIER THAN THE 1971 EDITION, SUMMER 1972 ADDENDA

Plant: North Anna Unit 1

Column 1	Column 2									C. 3
Material	Chemical Composition, Weight Percent									Notes
	C	Mn	P	S	Si	Cr	Ni	Mo	Cu	
BELTLINE MATERIALS										
Forging 05	0.20	0.71	0.013	0.012	0.21	0.39	0.74	0.64	0.16	(1)
Forging 04	0.21	0.75	0.010	0.019	0.21	0.33	0.82	0.64	0.12	
Forging 03	0.19	0.68	0.009	0.014	0.22	0.30	0.80	0.63	0.15	
Weld 05A	0.10	1.50	0.020(2)	ND	0.36	ND	0.10(2)	0.37	0.30	
Weld 05B	0.09	1.49	0.020(2)	ND	0.33	ND	0.10(2)	0.37	0.11	
Weld 04	0.06	1.29	0.020	0.012	0.35	0.25	0.11	0.49	0.086	
SURVEILLANCE MATERIALS										
Forging 03	0.20	0.68	0.019	0.011	0.26	0.30	0.79	0.61	0.16	(1)
Weld 04	0.06	1.29	0.020	0.012	0.35	0.025	0.[illegible]	0.49	0.086	

REQUIRED: State heat number of weld wires used for determining above chemical composition if different from that in ¶ (3). - Not applicable -

NOTES

(1) BAW-1911, Revision 1

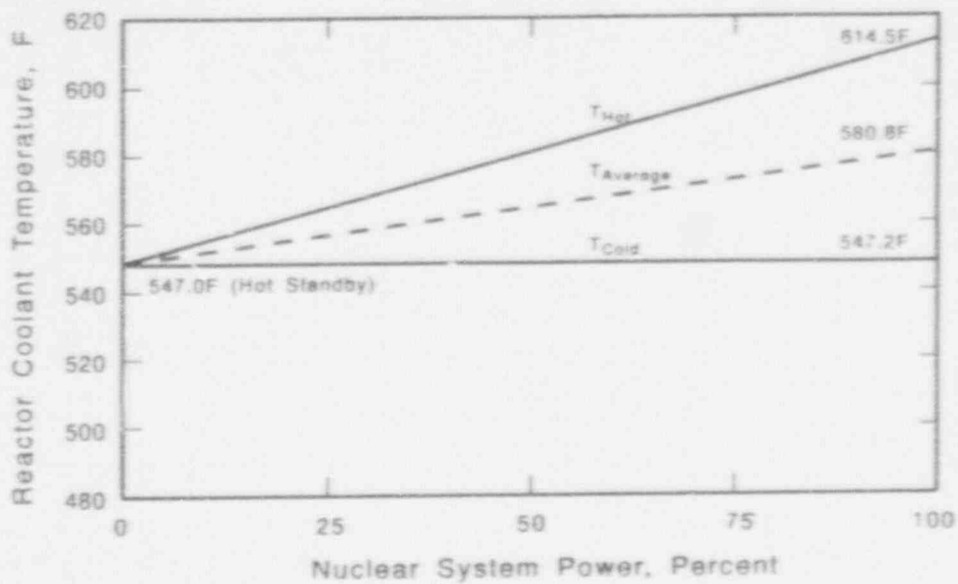
(2) Estimated value.

TABLE 8. GENERIC LETTER 92-01 RESPONSE: SECTION 3, ITEM a
Subject: Generic Letter 88-11 Response Commitments; Effect of Irradiation Temperature
Plant: North Anna Unit 1

Cold Leg Temperature (T_{cold}): 547.2 F

If T_{cold} is <525°F, state how this was considered in determination of embrittlement effects (C_vUSE, RT_{NDT}) in accordance with Regulatory Guide 1.99, Revision 2:

Not Applicable
(see Figure for current design operating temperatures)



References:

North Anna Units 1 and 2 Final Safety Analysis Report Volume 3, Docket No. 50-338, January 1973.

TABLE 9. GENERIC LETTER 92-01 RESPONSE: SECTION 3, ITEM b
Subject: Generic Letter 88-11 Response Commitments; Utilization of Surveillance Results
Plant: North Anna Unit 1
Were surveillance results used in determining C_vUSE? Yes ☐ No ✓
Were surveillance results used in determining RT_{NDT}? Yes ✓ No ☐
If any "yes" boxes were checked above, state how the surveillance results were used: The North Anna Unit 1 RVSP results from Capsule V and U were used to determine the adjusted reference temperature (ART) per Regulatory Guide 1.99, Revision 2, Position 2, for preparation of pressure-temperature limit curves.
References: BAW-2146

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 ΔRT_{NDT} exceed $\Delta RT_{NDT} + 2\sigma$ predicted by Regulatory Guide 1.99, Revision 2?						
Question II. Does measured C_vUSE 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 ΔRT_{NDT}	Predicted $\Delta RT_{NDT} + 2\sigma$	Question I Yes/No	Measured C_vUSE Drop	Predicted C_vUSE 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)	106	No	25(1)	15	Yes(4)
	65(2)	143	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 2.49×10^{18} nvt. This result was not observed for surveillance material irradiated to 8.28×10^{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_vUSE 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 3. GENERIC LETTER 92-01 RESPONSE: SECTION 2, ITEM b, ¶ (1)								
Subject: 10CFR50.61 and 10CFR50, Appendix G, III.A; Material Properties Related to PTS and Fracture Toughness Requirements								
Plant: North Anna Unit 2								
Column 1	Column 2				Column 3	Column 4	Column 5	C.6
Beltline Materials	Unirradiated Charpy Test Results				Unirrad. Dropwt. Test Results T_{NDT} F	Unirrad. RT_{NDT} F	Method of Determng RT_{NDT}	Notes
	Col. 2a	Col. 2b	Col. 2c	Col. 2d				
	C_v 10 F ft-lb	C_v 30 ft-lb F	C_v 50 ft-lb F	C_v 35 MLE F				
<u>FORGING</u>								
05	ND	ND	ND	ND	5	9	Estimated	(1)
04	ND	ND	ND	ND	-49	75	NB-2331	(1)
03	ND	ND	ND	ND	-13	56	NB-2331	(1)
<u>WELD</u>								
W05A	ND	ND	ND	ND	0	0	Estimated	(1)
W05B	ND	ND	ND	ND	0	0	Estimated	(1)
W04	ND	ND	ND	ND	-67	-48	NB-2331	(1)

NOTES:

(1) BAW-1911, Revision 1

TABLE 4. GENERIC LETTER 92-01 RESPONSE: SECTION 2, ITEM b, ¶ (2)		
Subject: 10CFR50.61 and 10CFR50, Appendix G, III.A; Material Properties Related to PTS and Fracture Toughness Requirements -- APPLICABLE ONLY TO REACTOR VESSELS CONSTRUCTED TO AN ASME CODE EARLIER THAN THE 1971 EDITION, SUMMER 1972 ADDENDA		
Plant: North Anna Unit 2		
Column 1	Column 2	Col. 3
Material	Heat Treatment	Notes
BELTLINE MATERIALS		
Forging 05	1688-1697F-2½hr/WQ; 1220-1229F-6hr/FC; 1130±25F-14¾hr/FC	(1,3)
Forging 04	1688-1697F-2½hr/WQ; 1220-1229F-6hr/FC; 1130±25F-14¾hr/FC	
Forging 03	1688-1697F-2½hr/WQ; 1220-1229F-6hr/FC; 1130±25F-14¾hr/FC	
Weld 05A	1130±25F-13½hr/FC	
Weld 05B	1130±25F-13½hr/FC	
Weld 04	1130±25F-13½hr/FC	
SURVEILLANCE MATERIALS		
Forging 04	1688-1697F-2½hr/WQ; 1220-1229F-6hr/FC; 1130±25F-14¾hr/FC	(2,3)
Weld W04	1130±25F-13½hr/FC	

NOTES:

(1) Estimated based on review of surveillance data.
(2) BAW-1911, Revision 1
(3) WQ - water quench
FC - furnace cool

TABLE 1. GENERIC LETTER 92-01 RESPONSE: SECTION 1	
Subject: 10CFR50, Appendix H; Adherence to RVSP Requirements	
Plant: North Anna Unit 2	
Question I: Does RVSP meet ASTM E 185-73, E 185-79, or E 185-82?	Yes ✓ No ☐
Question II: Is plant one of the following? ANO-1, Crystal River-3, Davis Besse, R. E. Ginna Oconee-1, Oconee-2, Oconee-3, Point Beach-1, Point Beach-2, Rancho Seco, Surry-1, Surry-2, Turkey Point-3, Turkey Point-4, Zion-1, Zion-2.	Yes ☐ No ✓
IF ANSWER IS "YES" TO EITHER QUESTION I OR QUESTION II, PROCEED TO TABLE 2.	
IF ANSWER IS "NO" TO BOTH QUESTION I AND QUESTION II, PROCEED TO QUESTION III AND QUESTION IV.	
Question III: If plan is to revise RVSP to meet requirements of 10CFR50, Appendix H, when will revised RVSP be submitted to NRC?	
Response: Not Applicable (See Question I and II above)	
Question IV: If plan is <u>not</u> to revise RVSP to meet requirements of 10CFR50, Appendix H, when will exemption from 10CFR50, Appendix H be requested from NRC?	
Response: Not Applicable (See Question I and II above)	

NOTES: (1) WCAP-8772, RVSP per ASTM E185-73

TABLE 2. GENERIC LETTER 92-01 RESPONSE: SECTION 2, ITEM a					
Subject: 10CFR50, Appendix G, C_vUSE Requirements					
Plant: North Anna Unit 2					
Column 1	Column 2		Column 3		Column 4
Limiting Material	Initial USE ft-lb	EFPY to reach C_vUSE<50 ft-lb	If Column 2 is within license period: C_vUSE at indicated time		Action taken per IV.A.1
			Column 3A	Column 3B	
			12/16/91	EOL	
<u>LIMITING BELTLINE WELD</u>					
Weld 04	107	>32	NA	NA	NA
<u>LIMITING BELTLINE FORGING</u>					
Forging 04	74	>32	NA	NA	NA

NOTES:

(1) Fluence values for C_vUSE taken at ¼-thickness.

NORTH ANNA UNIT 2

TABLE 5. GENERIC LETTER 92-01 RESPONSE: SECTION 2, ITEM b, ¶ (3)

Subject: 10CFR50.61 and 10CFR50, Appendix G, III.A; Material Properties Related to PTS and Fracture Toughness Requirements -- <u>APPLICABLE ONLY TO REACTOR VESSELS CONSTRUCTED TO AN ASME CODE EARLIER THAN THE 1971 EDITION, SUMMER 1972 ADDENDA</u>

Plant: North Anna Unit 2

Column 1	Column 2	Column 3	Column 4	Column 5	C. 6
Beltline Plate or Forging	Heat Number	Beltline Weld	Weld Wire Heat	Weld Flux Lot	Notes
Forging 05	990598/291396	Weld 05A	4278 (2)	1211 (4)	(1)
Forging 04	990496/292424	Weld 05B	801 (2)	1211 (4)	
Forging 03	990533/297355	Weld 04	716126 (3)	26 (5)	

NOTES:

(1) BAW-1911, Revision 1

(2) S4 Mo

(3) S3 Mo

(4) SMIT 89

(5) LW 320

TABLE 6. GENERIC LETTER 92-01 RESPONSE: SECTION 2, ITEM b, ¶ (4)				
Subject: 10CFR50.61 and 10CFR50, Appendix G, III.A: Material Properties Related to PTS and Fracture Toughness Requirements -- APPLICABLE ONLY TO REACTOR VESSELS CONSTRUCTED TO AN ASME CODE EARLIER THAN THE 1971 EDITION, SUMMER 1972 ADDENDA				
Plant: North Anna Unit 2				
Column 1	Column 2	Column 3	Column 4	Column 5
Surveillance Plate or Forging Heat Number	Surveillance Weld	Weld Wire Heat	Weld Flux Lot	Notes
Forging 04 990496/292424	Weld 04	716126 (S3 Mo)	26 (LW 320)	(1)

NOTES:

(1) BAW-1911, Revision 1

TABLE 7. GENERIC LETTER 92-01 RESPONSE: SECTION 2, ITEM b, ¶ (5)

Subject: 10CFR50.61 and 10CFR50, Appendix G, III.A; Material Properties Related to PTS and Fracture Toughness Requirements -- APPLICABLE ONLY TO REACTOR VESSELS CONSTRUCTED TO AN ASME CODE EARLIER THAN THE 1971 EDITION, SUMMER 1972 ADDENDA

Plant: North Anna Unit 2

Column 1	Column 2									C. 3
Material	Chemical Composition, Weight Percent									Notes
	C	Mn	P	S	Si	Cr	Ni	Mo	Cu	
BELTLINE MATERIALS										
Forging 05	0.20	0.68	0.010	0.013	0.25	0.34	0.77	0.60	0.08	(1)
Forging 04	0.195	0.78	0.011	0.016	0.24	0.35	0.83	0.62	0.09	
Forging 03	0.16	0.66	0.013	0.017	0.15	0.34	0.93	0.59	0.13	
Weld 05A	0.09	1.49	ND	ND	0.33	ND	0.10 (2)	0.37	0.11	
Weld 05B	0.086	1.58	0.012	0.012	0.43	ND	0.10 (2)	0.51	0.18	
Weld 04	0.08	1.82	0.017	0.011	0.25	0.042	0.084	0.49	0.088	
SURVEILLANCE MATERIALS										
Forging 04	0.19	0.76	0.018	0.011	0.25	0.35	0.86	0.60	0.11	(1)
Weld 04	0.08	1.82	0.017	0.011	0.25	0.042	0.084	0.49	0.088	

REQUIRED: State heat number of weld wires used for determining above chemical composition if different from that in ¶ (3). - Not applicable -

NOTES

(1) BAW-1911, Revision 1

(2) Estimated value

TABLE 8. GENERIC LETTER 92-01 RESPONSE: SECTION 3, ITEM a

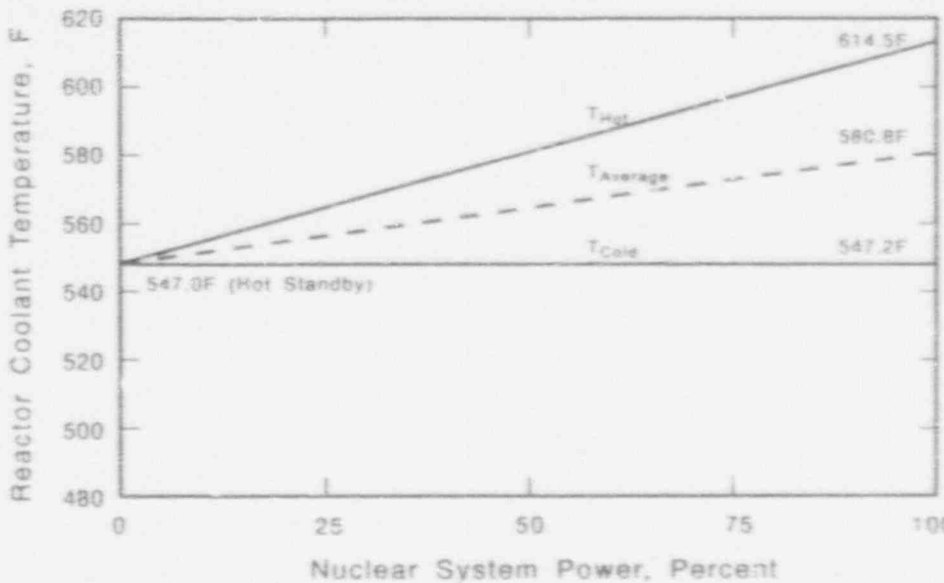
Subject: Generic Letter 88-11 Response Commitments; Effect of Irradiation Temperature

Plant: North Anna Unit 2

Cold Leg Temperature (T_{cold}): 547.2 F

If T_{cold} is <525°F, state how this was considered in determination of embrittlement effects (C_vUSE, RT_{NDT}) in accordance with Regulatory Guide 1.99, Revision 2:

Not Applicable
(see Figure for current design operating temperatures)



References:

North Anna Units 1 and 2 Final Safety Analysis Report Volume 3, Docket No. 50-339, January 1973.

TABLE 9. GENERIC LETTER 92-01 RESPONSE: SECTION 3, ITEM b

Subject: Generic Letter 88-11 Response Commitments; Utilization of Surveillance Results

Plant: North Anna Unit 2

Were surveillance results used in determining C_vUSE? Yes ☐ No ✓

Were surveillance results used in determining RT_{NDT}? Yes ✓ No ☐

If any "yes" boxes were checked above, state how the surveillance results were used:

The North Anna Unit 2 RVSP results from Capsule V and U were used to determine the adjusted reference temperature (ART) per Regulatory Guide 1.99, Revision 2, Position 2, for preparation of pressure-temperature limit curves.

References:

BAW-2146

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 ΔRT_{NDT} exceed ΔRT_{NDT} + 2σ predicted by Regulatory Guide 1.99, Revision 2?						
Question II. Does measured C_vUSE 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 ΔRT_{NDT}	Predicted ΔRT_{NDT} + 2σ	Question I Yes/No	Measured C_vUSE Drop	Predicted C_vUSE Drop	Question II Yes/No
Forging 05	ND	ND	NA	ND	ND	NA
Forging 04	9(1)	70	No	13(1)	9	Yes(3)
	60(2)	91	No	0(2)	12	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_vUSE 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.

5. REFERENCES

1. A. L. Lowe, Jr., "Reactor Pressure Vessel and Surveillance Program Materials Licensing Information for North Anna Units 1 and 2," BAW-1911, Revision 1, Babcock & Wilcox, Lynchburg, Virginia, August 1986.
2. A. D. Nana and M. J. DeVan, "North Anna Unit 1 Pressure-Temperature Limits for 12 EFPY and North Anna Unit 2 Pressure-Temperature Limits for 12 and 15 EFPY," BAW-2146, B&W Nuclear Service Company, Lynchburg, Virginia, October 1991.
3. 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.
4. S. E. Yanichko, L. Albertin, and E. P. Lippincott, "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.
5. A. L. Lowe, Jr., "Analysis of Capsule V Virginia Electric & Power Company North Anna Unit No. 2 Reactor Vessel Materials Surveillance Program," BAW-1794, Babcock & Wilcox, Lynchburg, Virginia, October 1983.
6. 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. Final Safety Analysis Report, Volume 3, North Anna Power Station, Units 1 and 2, Docket No. 50-338, January 1973.

8. Final Safety Analysis Report, Volume 3, North Anna Power Station, Units 1 and 2, Docket No. 50-339, January 1973.

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. 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.

6. CERTIFICATION

This report accurately responds to the request for information stated in Generic Letter 92-01.

M. J. DeVan 6/15/92
M. J. DeVan, Engineer II Date
Materials and Structural Analysis Unit

A. L. Lowe, Jr. 6/15/92
A. L. Lowe, Jr., Advisory Eng. Date
Materials and Structural Analysis Unit

This report was reviewed and found to be accurate.

L. B. Gross 6/15/92
L. B. Gross, Advisory Eng. Date
Materials and Structural Analysis Unit

Verification of independent review.

K. E. Moore 6-15-92
K. E. Moore, Manager Date
Materials and Structural Analysis Unit

This report is approved for release.

A. W. Robinson 6/15/92
A. W. Robinson, Project Manager Date
Owners Group Engineering Services