VIRGINIA ELECTRIC AND POWER COMPANY
RICHMOND, VIRGINIA 23261

June 22, 2000

U.S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, D.C. 20555

Serial No.	00-306
NL&OS/GSS/ETS	R0
Docket Nos.	50-338
	50-339
License Nos.	NPF-4
	NPF-7

Gentlemen:

VIRGINIA ELECTRIC AND POWER COMPANY
NORTH ANNA POWER STATION UNITS 1 AND 2
PROPOSED TECHNICAL SPECIFICATIONS CHANGES
REQUESTS FOR EXEMPTION PER 10 CFR 50.60(b)
REACTOR COOLANT SYSTEM PRESSURE/TEMPERATURE LIMITS
LTOPS SETPOINTS, AND LTOPS ENABLE TEMPERATURES

By letter dated November 19, 1999, Virginia Electric and Power Company (Virginia Power) transmitted to the NRC a detailed evaluation of available reactor vessel materials surveillance data, including data derived from the recently-analyzed North Anna Unit 1 Capsule W. The evaluation demonstrated that North Anna Units 1 and 2 continue to meet the 10 CFR 50.61 Pressurized Thermal Shock screening criteria at cumulative core burnups up to 32.3 Effective Full Power Years (EFPY) and 34.3 EFPY (corresponding to end-of-license) for Units 1 and 2, respectively. However, the cumulative core burnup limit for the existing North Anna Unit 1 Technical Specification Reactor Coolant System (RCS) pressure/temperature (P/T) operating limits, Low Temperature Overpressure Protection System (LTOPS) setpoints, and LTOPS enable temperature (T_{enable}) values were determined to be no longer valid. Therefore a licensing submittal with revised North Anna Unit 1 Technical Specification RCS P/T limits, LTOPS setpoints, and T_{enable} value is necessary.

Although the November 19, 1999 submittal demonstrated that the existing North Anna Unit 2 RCS P/T limits, LTOPS setpoints, and T_{enable} values continue to be valid and conservative, North Anna Unit 2 is predicted to reach the cumulative core burnup applicability limit of 17 EFPY in September 2001. Therefore, in order to extend the cumulative core burnup applicability limit, and to maintain consistent analytical bases for Units 1 and 2, the North Anna Unit 2 Technical Specification P/T limits, LTOPS setpoints, and LTOPS T_{enable} are also re-evaluated herein.

Thus, Virginia Power requests amendments, in the form of changes to the Technical Specifications to Facility Operating Licenses Numbers NPF-4 and NPF-7 for North Anna Power Station Units 1 and 2, respectively to extend the cumulative core burnup

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applicability llimits for the Technical Specification RCS P/T limits, LTOPS setpoints, and T_{enable} values. The proposed changes are discussed in Attachment 1. The proposed Technical Specifications changes are provided as a mark-up in Attachment 2 and a typed version in Attachment 3.

Pursuant to 10 CFR 50.60(b), the revised analysis bases require exemptions from the requirements of 10 CFR 50 Appendix G to permit application of ASME Section XI Code Case N-640 to North Anna Units 1 and 2. In addition, an exemption to permit a plant-specific application of the analysis methodology that supports ASME Section XI Code Case N-514 to North Anna Units 1 and 2 is also required. The proposed bases of the revised analysis demonstrate the conservatism of the existing North Anna Units 1 and 2 Technical Specification P/T limit curves, LTOPS setpoints, LTOPS T_{enable} values, and associated equipment operability requirements established to maintain the validity of the LTOPS design basis accident analyses. The proposed exemptions and the basis for the exemptions are included in Attachment 1.

We have evaluated the proposed changes and have determined that they do not involve a significant hazards consideration as defined in 10 CFR 50.92. The basis for our determination that the changes do not involve a significant hazards consideration is provided in Attachment 4. We have also determined that operation with the proposed changes will not result in any significant increases in the amounts of effluents that may be released offsite and in any significant increases in individual or cumulative occupational radiation exposure. Therefore, the proposed amendment is eligible for categorical exclusion as set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment is needed in connection with the approval of the proposed changes.

In our letter of November 19, 1999, we concluded that the limiting North Anna Unit 1 reactor vessel beltline material (lower shell forging) would exceed the design basis RT_{NDT} value of 169.2°F at 17.2 EFPY. This cumulative core burnup is predicted to be reached in May 2001. Unit 2 is predicted to reach the cumulative core burnup applicability limit for the Technical Specification P/T limits, LTOPS setpoints, and LTOPS T_{enable} valve in September 2001. Therefore, Virginia Power requests NRC approval of the proposed Technical Specification changes and exemption requests by February 2001. Implementation of these changes will be within 30 days of date of the amendments to the Technical Specifications.

By letter dated September 10, 1999, Virginia Power transmitted the North Anna Unit 1 reactor vessel materials surveillance program Capsule W analysis results to the NRC. The results were documented in BAW-2356 Revision 0. Subsequent to the issuance of BAW-2356 Revision 0, several non-technical errors in the report were identified. To correct these errors, BAW-2356 Revision 1 includes: (a) modifications to the text presented in Section 6.1 (introduction to the neutron fluence analysis methodology), and (b) revised signatures in Section 9. Neither of these changes affects the results or conclusions of the surveillance capsule analysis report, or of the evaluation of the capsule analysis results transmitted to the NRC by letter dated November 19, 1999. BAW-2356 Revision 1 is included as Attachment 5 for your information.

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

Very truly yours,

David A. Christian
Senior Vice President and
Chief Nuclear Officer

Attachments:

Attachment 1	Discussion of Changes
Attachment 2	Mark-up of Technical Specifications Changes
Attachment 3	Proposed Technical Specifications Changes
Attachment 4	Significant Hazards Consideration Determination
Attachment 5	Technical Report BAW-2356, Revision 1

Commitments made in this letter:

1. There are no commitments in this letter

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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 David A. Christian who is Senior Vice President and Chief Nuclear Officer of Virginia Electric and Power Company. He has affirmed before me that he is duly authorized to execute and file the foregoing document in behalf of that Company, and that the statements in the document are true to the best of his knowledge and belief.

Acknowledged before me this 22nd day of June, 2000.

My Commission Expires: 3/31/04.

Maggie McClure
Notary Public

(SEAL)

Attachment 1

Discussion of Changes

North Anna Power Station
Units 1 and 2
Virginia Electric and Power Company

Table of Contents

1.0 Introduction

As a result of the North Anna Unit 1 reactor vessel materials surveillance capsule W analysis, the cumulative core burnup applicability limit for the existing North Anna Unit 1 Technical Specification Reactor Coolant System (RCS) Pressure/Temperature (P/T) operating limits, Low Temperature Overpressure Protection System (LTOPS) power operated relief valve (PORV) lift setpoints, and LTOPS enable temperature (T_{enable}) has been determined to be no longer valid. The existing North Anna Unit 2 P/T limits, LTOPS setpoints, and T_{enable} are valid to a cumulative core burnup of 17 EFPY, which is predicted to be reached in September 2001. Therefore, Virginia Power proposes amendments to the North Anna Units 1 and 2 Technical Specifications to extend the cumulative core burnup applicability limits for the P/T limits, LTOPS setpoints, and T_{enable} values.

The proposed extension of the cumulative core burnup applicability limits is accomplished by revising the design basis P/T limit curves. The proposed revised design basis P/T limit curves utilize ASME Section XI Code Case N-640, which supports use of a conservative but less restrictive stress intensity formulation (K_{1c}). Therefore, the proposed revised design basis P/T limit curves are significantly less limiting than the existing Technical Specification P/T limit curves. The evaluation presented herein demonstrates that the existing Technical Specification P/T limit curves and LTOPS setpoints remain conservative for the proposed extended cumulative core burnup applicability limit, and need not be changed.

The existing Technical Specification LTOPS T_{enable} values are also demonstrated herein to remain conservative for the proposed extended cumulative core applicability limit. This is accomplished by a plant-specific application of the analysis methodology that supports ASME Section XI Code Case N-514.

Implementation of the revised cumulative core burnup applicability limits requires changes to the North Anna Units 1 and 2 Technical Specifications. Use of ASME Section XI Code Case N-640 and plant-specific application of the analysis methodology that supports Code Case N-514 require exemptions from the requirements of 10 CFR 50 Appendix G. A discussion of the proposed changes and their safety significance are presented below.

2.0 Background

Following the analysis of North Anna Unit 1 capsule W [1], Virginia Power performed a detailed evaluation of available reactor vessel materials surveillance data. This information was transmitted to the NRC by Reference [2]. As documented in Reference [2], the PTS screening calculation results for North Anna Units 1 and 2 were determined to meet the applicable screening criteria for cumulative core burnups up to 32.3 Effective Full Power Years (EFPY) and 34.3 EFPY (corresponding to end-of-license) for Units 1 and 2, respectively. However, the cumulative core burnup limit for the current North Anna Unit 1 RCS P/T limits, LTOPS setpoints, and LTOPS enable temperature (T_{enable}) values documented in the North Anna Technical Specifications was determined to be no longer valid. Specifically, the newly acquired North Anna Unit 1 surveillance data caused the cumulative core burnup limit for the Unit 1 RCS P/T limit curves to be reduced from 30.7 EFPY to 17.2 EFPY. (The cumulative core burnup limit of 17.0 EFPY for the currently applicable North Anna Unit 2 P/T limit curves was determined to remain valid.) North Anna Unit 1 is predicted to achieve 17.2 EFPY in May 2001. Therefore, Virginia Power committed to provide a licensing submittal with revised North Anna Unit 1 Technical Specification P/T limits, LTOPS setpoints, and T_{enable} value by June 30, 2000.

As described in Reference [2], the existing North Anna Unit 1 P/T limits and LTOPS setpoints are based on a limiting ¼-thickness (¼-T) RT_{NDT} of 162.9°F [3][4]. When the P/T limits and LTOPS setpoints were developed, this value of RT_{NDT} was determined to bound all North Anna Unit 1 reactor vessel beltline materials at end-of-license (EOL) reactor vessel beltline fluences corresponding to 30.7 EFPY [3][4]. After consideration of the changes to previously reported information described in Reference [2], the most limiting ¼-T RT_{NDT} value for North Anna Unit 1 was determined to be 174.9°F, which exceeds the ¼-T RT_{NDT} value assumed in the existing Unit 1 P/T limits, LTOPS setpoints, and LTOPS T_{enable} [3][4]. The 174.9°F value of RT_{NDT} was determined on the basis of fluence values corresponding to an end-of-license cumulative core burnup of 32.3 EFPY [5]. Virginia Power calculations demonstrate that the limiting North Anna Unit 1 reactor vessel beltline material (lower shell forging) will exceed the design basis ¼-T RT_{NDT} value of 162.9°F at 17.2 EFPY, which is predicted to be reached in May 2001.

The existing North Anna Unit 2 P/T limits and LTOPS setpoints [3][4] were also evaluated in Reference [2]. The North Anna 2 P/T limits are based on a limiting ¼-T RT_{NDT} of 196.5°F [3] [4]. When the P/T limits and LTOPS setpoints were developed, this value of RT_{NDT} was determined to bound all North Anna Unit 2 reactor vessel beltline materials at reactor vessel beltline material fluences corresponding to 17.0 EFPY [3] [4]. After consideration of the changes to previously reported information described in Reference [2], the most limiting ¼-T RT_{NDT} value for North Anna Unit 2 was determined to be 209.4°F at a fluence corresponding to an end-of-license cumulative core burnup of 34.3 EFPY [5]. Virginia Power calculations demonstrate that the ¼-T RT_{NDT} value for the limiting North Anna Unit 2 reactor vessel beltline material (lower shell forging) at a fluence corresponding to 17.0 EFPY [5] is 193.1°F. Therefore, the existing North Anna Unit 2 RCS P/T limits and LTOPS setpoints [3] [4] were determined to remain valid and conservative. However, North Anna Unit 2 is predicted to reach 17 EFPY in September 2001 [2]. Therefore, in order to extend the cumulative core burnup applicability limit, and to maintain consistent analytical bases for Units 1 and 2, the North Anna Unit 2 Technical Specification P/T limits, LTOPS setpoints, and LTOPS T_{enable} are also being re-evaluated herein.

3.0 Licensing and Design Bases

The existing North Anna Units 1 and 2 Technical Specification P/T limits, LTOPS setpoints, and T_{enable} values are based on limiting ¼-thickness (¼-T) RT_{NDT} values of 162.9°F and 196.5°F, respectively [3] [4]. The P/T limits were developed assuming heatup rates of 20°F/hr, 40°F/hr, and 60°F/hr, and cooldown rates of 0°F/hr (steady-state), 20°F/hr, 40°F/hr, 60°F/hr, and 100°F/hr. The existing Technical Specification P/T limits include a correction for the pressure difference between the point of measurement (i.e., the pressurizer) and the point of interest (i.e., the reactor vessel beltline), including the effects of RCP operation. The P/T limits do not include instrumentation uncertainties on the basis that these uncertainties are insignificant when compared to the margin terms included in the ASME Section XI Appendix G methods (i.e., 2.0 multiplier on pressure stress). The criticality limit required by 10 CFR 50 Appendix G is not included in the existing Technical Specification P/T limit curves, since North Anna Units 1 and 2 Technical

Specification LCO 3.1.1.5 defines a minimum temperature for criticality that is substantially more limiting than that required by 10 CFR 50 Appendix G.

Conservative analyses of the design basis cold overpressurization events were performed in support of the existing Technical Specification LTOPS setpoints. The design basis cold overpressurization events are (a) mass addition due to inadvertent startup of a charging pump, and (b) heat addition due to startup of a reactor coolant pump with a 50°F temperature difference between the steam generator secondary and the RCS. Limitations on charging pump, high head safety injection pump, and reactor coolant pump operations were established to ensure that the assumptions of the design basis cold overpressurization event analyses remain valid. The LTOPS setpoints were designed to provided bounding protection against 100% of the isothermal P/T limit curve. The LTOPS enable temperature (T_{enable}) was established on the basis of ASME Section XI Working Group on Operating Plant Criteria (WGOPC) recommendations, which define T_{enable} as RT_{NDT} + 50°F. A bounding temperature measurement uncertainty was included in the proposed Technical Specification values for T_{enable}. Margin was not added to compensate for the maximum calculated temperature difference between the downcomer fluid and the ¼-T reactor vessel location. References [3] and [4] established that the design condition for the LTOPS design basis events is isothermal, which precludes the need for additional margin in T_{enable} to compensate for fluid/metal temperature differences due to finite heatup rates. An administrative limitation on heatup and cooldown rate of 50°F/hr was established for all operating Modes except Mode 6 (Refueling Shutdown).

Virginia Power proposes to replace the current design and licensing basis P/T limits, including the isothermal (steady-state) P/T limit curve that constitutes the design limit for the LTOPS setpoint analysis, with those documented in Appendix C [7]. Further, Virginia Power proposes to replace the current design and licensing basis RT_{NDT} calculations, and the associated relationship of cumulative core burnup to reactor vessel neutron fluence, with those previously submitted in Reference [2]. Finally, Virginia Power proposes to modify the analysis basis for the LTOPS T_{enable} values. Other features of the existing design, as described in the preceding paragraphs, remain unchanged.

4.0 Discussion of Changes

The cumulative core burnup applicability limits for the current North Anna Units 1 and 2 Technical Specification P/T limits, LTOPS setpoints, and LTOPS T_{enable} values will be reached at 17.2 EFPY and 17.0 EFPY, respectively. However, two sources of analytical margin are available to support extension of the cumulative core burnup applicability limits of the existing P/T limits and LTOPS setpoints. These two sources of margin are (a) ASME Section XI Code Case N-640 [6], which supports use of the K_{1c} fracture toughness curve, instead of the K_{1a} curve employed in the development of the existing P/T limits and LTOPS setpoints [3] [4], and (b) utilization of an alternate formulation for the LTOPS Enable Temperature (T_{enable}) based on a fracture criterion, instead of the generic ASME Section XI Code Case N-514 formulation employed in References [3] and [4]. Substitution of these alternate methodologies into the LTOPS and P/T limits design analyses provides sufficient margin to extend the cumulative core burnup applicability limits for the existing P/T limits and LTOPS setpoints to values corresponding to the end of the current license period. This approach greatly simplifies the implementation process for the revised P/T limits, LTOPS setpoint, and T_{enable} design analysis.

The existing North Anna Units 1 and 2 P/T limits were developed in References [9] and [10], respectively. For convenience, the Reference [9] and [10] P/T limits are reproduced in Appendices A and B, respectively. Revised North Anna Units 1 and 2 P/T limit curves for normal operation were developed in WCAP-15112, Revision 1 [7]. The portions of Reference [7] that are applicable to the present analysis have been excerpted, and are presented in Appendix C. The Appendix C curves were developed to support operation during a postulated 20-year license renewal period for North Anna Units 1 and 2. The ASME Section XI Code Case N-640 K_{Ic} formulation [6] and a limiting ¼-T RT_{NDT} value of 218.5°F were employed in the development of the Appendix C curves. The limiting ¼-T RT_{NDT} value was determined using then-available surveillance data and end-of-license-renewal reactor vessel beltline fluence values corresponding to 50.3 EFPY and 54.3 EFPY for North Anna Units 1 and 2, respectively. The relationship between cumulative core burnup and reactor vessel beltline fluence that was used to establish the cumulative core burnup limits for the Appendix C [7] P/T limits is based on the approved Virginia Power Reactor Vessel Fluence Analysis Methodology Topical Report [5]. This relationship was also used in the Reference [2] submittal. The evaluation documented herein demonstrates that the existing P/T limit curves [3] [4], which are based on the ASME Section XI Appendix G K_{Ia} formulation and a limiting value of RT_{NDT} of 162.9°F, conservatively bound the P/T limits curves presented in Appendix C, which are based on the ASME Section XI Appendix G K_{Ic} formulation and a limiting ¼-T RT_{NDT} value of 218.5°F. Thus, the cumulative core burnup applicability limit for the existing North Anna Units 1 and 2 Technical Specification P/T limit curves may be extended by simply revising the design and licensing basis P/T limit curves to be those presented in Appendix C [7].

A similar approach is taken to extend the cumulative core burnup applicability limit for the North Anna Units 1 and 2 Technical Specification LTOPS setpoints. The basis for the existing LTOPS setpoints is to provide conservative bounding protection for 100% of the isothermal P/T limit curve [3] [4]. The present analysis demonstrates that the existing isothermal P/T limit curve [3] [4], which was developed using the ASME Section XI Appendix G K_{Ia} formulation and a limiting value of RT_{NDT} of 162.9°F, conservatively bounds 100% of the proposed revised design basis isothermal P/T limit curve, which was developed using the ASME Section XI Appendix G K_{Ic} formulation [7] and a limiting ¼-T RT_{NDT} value of 218.5°F. In this manner, the existing North Anna Units 1 and 2 Technical Specification LTOPS setpoints are demonstrated to provide conservative bounding protection for 100% of the proposed revised design basis isothermal limit curve presented in Appendix C [7] at an extended cumulative core burnup applicability limit.

The current design and licensing basis LTOPS enable temperature (T_{enable}) [3] [4] is based on recommendations of the ASME Section XI Working Group on Operating Plant Criteria (eventually codified as ASME Section XI Code Case N-514 [8]), which require LTOPS to be effective at coolant temperatures less than 200°F or at coolant temperatures corresponding to a reactor vessel metal temperature less than RT_{NDT} + 50°F. The existing North Anna Units 1 and 2 Technical Specification T_{enable} values were therefore calculated as RT_{NDT} + 50°F + temperature measurement uncertainty. Margin was not added to compensate for the maximum calculated temperature difference between the downcomer fluid and the ¼-T reactor vessel location. References [3] and [4] established that the design condition for the LTOPS design basis events is isothermal, which precludes the need for additional margin in T_{enable} to compensate for fluid/metal temperature differences due to finite heatup rates. The generic guidance for establishing T_{enable} presented in ASME Section XI Code Case N-514 is fundamentally based on a conservative assessment of margin to vessel fracture (i.e., a fracture criterion). An alternate T_{enable} methodology approved by the ASME Section XI Subcommittee on Nuclear Inservice Inspection, but not yet published, provides a means of calculating a conservative, plant-specific value of T_{enable} based on a fracture criterion. This alternate LTOPS T_{enable} methodology provides sufficient margin to extend the cumulative core burnup applicability limits for the existing North Anna Units 1 and 2 Technical Specification T_{enable} values, despite the increased design basis RT_{NDT} value identified in Reference [2]. The revised North Anna Units 1 and 2 T_{enable} analysis bases are documented in Appendix D.

4.1 Reactor Vessel Fluence (E > 1 MeV) versus Cumulative Core Burnup

The tables below the present reactor vessel neutron fluence (E>1 MeV) as a function of cumulative core burnup. This information was developed in accordance with the NRC-approved Virginia Power Reactor Vessel Fluence Analysis Methodology Topical Report [5]. The end-of-license (EOL) EFPY/fluence values presented below were used in the calculations supporting the Reference [2] submittal, which established revised design values of RT_{NDT} for North Anna Units 1 and 2.

Summary of Fluence Values Used to Calculate the North Anna Units 1 and 2 Limiting RT_{NDT} Values

EFPY	Peak Clad / Base Metal Fluence (n/cm^2, E > 1.0 MeV)	¼-T Fluence (n/cm^2, E > 1.0 MeV)	¾-T Fluence (n/cm^2, E > 1.0 MeV)
Unit 1			
32.3 (EOL)	3.92×10^{19}	2.446×10^{19}	0.952×10^{19}
50.3 (EOLR)	5.90×10^{19}	3.681×10^{19}	1.433×10^{19}
Unit 2			
34.3 (EOL)	3.96×10^{19}	2.471×10^{19}	0.962×10^{19}
54.3 (EOLR)	5.91×10^{19}	3.687×10^{19}	1.435×10^{19}

4.2 RT_{NDT} versus Reactor Vessel Fluence (E > 1 MeV)

The most recent evaluation of available reactor vessel material properties data for North Anna Units 1 and 2 [2] was transmitted to the NRC by Reference [2]. The most limiting 1/4-T RT_{NDT} value for North Anna Unit 1 was determined to be 174.9°F, which was based on an end-of-license fluence of 3.92×10^{19} n/cm^2 predicted to occur at a cumulative core burnup of 32.3 EFPY [5]. The most limiting ¼-T RT_{NDT} value for North Anna Unit 2 was determined to be 209.4°F, which was based on an end-of-license fluence of 3.96×10^{19} n/cm^2 predicted to occur at a cumulative core burnup of 34.3 EFPY [5]. The P/T limit curves presented in Appendix C [7] were developed assuming a design ¼-T RT_{NDT} value of 218.5°F. This value conservatively bounds RT_{NDT} values for North Anna Units 1 and 2 reactor vessel beltline materials at reactor vessel beltline fluence values corresponding to cumulative core burnups of 32.3 EFPY and 34.3 EFPY for Units 1 and 2, respectively.

4.3 Pressure/Temperature Limit Curves

Revised reactor coolant system (RCS) pressure/temperature limit curves based on the ASME Section XI Code Case N-640 K_{1c} formulation [6], a limiting ¼-T RT_{NDT} value of 218.5°F, and a limiting ¾-T RT_{NDT} value of 195.6°F were developed in Reference [7]. The portions of the Reference [7] analysis applicable to the present evaluation are documented in Appendix C.

4.4 Reactor Vessel Operating and Dimensional Data

The following dimensional data applies to the North Anna Units 1 and 2 reactor vessels [7]:

p = reactor vessel design presure = 2500 psia
R_I = vessel inner radius (in.) = 78.95 in.
t = vessel wall thickness (in.) = 7.705 in.

4.5 Conservatism of the Current Technical Specification P/T Limits for North Anna Units 1 and 2

The currently applicable North Anna Units 1 and 2 Technical Specification P/T limit curves are based on the analyses documented in References [9] and [10], respectively. Appendices A and B present the Reference [9] and [10] P/T limit curves, without correction for the pressure difference between the point of measurement (i.e., the pressurizer) and the point of interest (i.e., the reactor vessel beltline), to permit direct comparison with the proposed revised, and unmodified, design basis P/T limit curves presented in Appendix C. By inspection, it is evident that the proposed revised design basis P/T limit curves are conservatively bounded by the P/T limit curves upon which the existing North Anna Units 1 and 2 Technical Specification P/T limit curves are based. Because the proposed revised design basis P/T limit curves are based on a ¼-T RT_{NDT} value of 218.5°F, which conservatively bounds the most limiting ¼-T RT_{NDT} value at cumulative core burnups of 32.3 EFPY and 34.3 EFPY for North Anna Units 1 and 2 (as documented in Reference [2]), the existing North Anna Units 1 and 2 Technical Specification P/T limit curves are concluded to remain conservative for North Anna Units 1 and 2 cumulative core burnups up to 32.3 EFPY and 34.3 EFPY.

The 10 CFR 50 Appendix G criticality limit lines presented in Figures 1 through 3 of Appendix C are bounded by the more restrictive Technical Specification 3.1.1.5 Minimum Temperature for Criticality. Therefore, the 10 CFR 50 Appendix G criticality limit requirement is satisfied for North Anna Units 1 and 2 cumulative core burnups up to 32.3 EFPY and 34.3 EFPY. Finally, the hydrostatic test limitations determined in the Appendix C analysis [7] are conservatively bounded by those of References [9] and [10]. Therefore, the existing North Anna Units 1 and 2 Technical Specification hydrostatic test limits remain valid for cumulative core burnups up to 32.3 EFPY and 34.3 EFPY, respectively.

4.6 Conservatism of the Current Technical Specification LTOPS Setpoints for North Anna Units 1 and 2

The currently applicable North Anna Units 1 and 2 Technical Specification LTOPS setpoints were designed to provide bounding protection against 100% of the isothermal P/T limit curves presented in References [9] and [10] for Units 1 and 2, respectively. Again, it is evident that the proposed revised design basis isothermal P/T limit curve is conservatively bounded by the isothermal P/T limit curves upon which the existing North Anna Units 1 and 2 Technical Specification LTOPS setpoints are based. Because the proposed revised design basis P/T limit curves are based on a ¼-T RT_{NDT} value of 218.5°F, which conservatively bounds the most limiting ¼-T RT_{NDT} value at cumulative core burnups of 32.3 EFPY and 34.3 EFPY for North Anna Units 1 and 2 (as documented in Reference [2]), the existing North Anna Units 1 and 2 LTOPS setpoints are concluded to remain conservative for North Anna Units 1 and 2 cumulative core burnups up to 32.3 EFPY and 34.3 EFPY.

4.7 Conservatism of the Current Technical Specification T_{enable} Values for North Anna Units 1 and 2

Using the reactor vessel operating and dimensional data presented above, and the alternate T_{enable} methodolgy presented in Appendix D, the following T-enable value function is calculated:

$$T = RT_{NDT} + 50 \ln [((1.1 \bullet M_m (p R_I /t)) - 33.2)/20.734]$$ Appendix D, Equation (13)

$M_m = 0.926\, t^{1/2}$ for IS axial flaw, $2 \le t^{1/2} \le 3.464$
p = vessel design pressure = 2.5 ksia
R_I = 78.95 in.
t = 7.705 in.

$$T = RT_{NDT} + 50 \ln [(1.1 \bullet 0.926 \bullet 7.705^{1/2} \bullet (2.5 \bullet 78.95/7.705) - 33.2)/20.734]$$

$$T = RT_{NDT} + 31.9°F$$

(Note: The ASME Section XI formulation for the membrane stress correction factor, M_m is valid, since $t^{1/2} = (7.705)^{1/2} = 2.78$, which satisfies the inequality $2 \le t^{1/2} \le 3.464$.)

The limiting North Anna Unit 1 ¼-T RT_{NDT} value of 174.9°F (lower shell forging) is predicted to occur at an end-of-license cumulative core burnup of 32.3 EFPY [2]. After consideration of a bounding value of temperature measurement uncertainty of 20°F, a revised a T_{enable} value of 226.8°F is determined to be applicable to North Anna Unit 1 operation to 32.3 EFPY. This value is conservatively bounded by the T_{enable} value of 235°F in the currently applicable North Anna Unit 1 Technical Specifications. This value also conservatively bounds the threshold temperature at which limitations on charging pump, high head safety injection pump, and reactor coolant pump operations are established to ensure that the assumptions of the design basis cold overpressurization event analyses remain valid.

The limiting North Anna Unit 2 ¼-T RT_{NDT} value of 209.4°F (lower shell forging) is predicted to occur at an end-of-license cumulative core burnup of 34.3 EFPY [2]. After consideration of a bounding value of temperature measurement uncertainty of 20°F, a revised a T_{enable} value of 261.3°F is determined to be applicable to North Anna Unit 2 operation to 34.3 EFPY. This value is conservatively bounded by the T_{enable} value of 270°F in the currently applicable North Anna Unit 2 Technical Specifications. This value also conservatively bounds the threshold temperature at which limitations on charging pump, high head safety injection pump, and reactor coolant pump operations are established to ensure that the assumptions of the design basis cold overpressurization event analyses remain valid.

5.0 Specific Changes to North Anna Units 1 and 2 Technical Specifications

The following specific changes to the North Anna Units 1 and 2 Technical Specifications are proposed:

Figures 3.4-2 and 3.4-3, Reactor Coolant System Heatup and Cooldown Limitations: The "material property basis" and the cumulative core burnup applicability limits on North Anna Units 1 and 2 Figures 3.4-2 and 3.4-3 are being modified to reflect the revised design analysis for the reactor coolant system pressure/temperature operating limits.

Bases for Section 3/4.4.9, "Pressure/Temperature Limits, Reactor Coolant System" and "Pressure/Temperature Limits, Low-Temperature Overpressure Protection": The bases for Section 3/4.4.9 is being modified to reflect the revised design analysis for the pressure/temperature operating limits, LTOPS setpoints, and LTOPS enable temperatures.

6.0 Basis for Exemption from the Requirements of 10 CFR 50 Appendix G

The evaluations which support the proposed changes to the North Anna Units 1 and 2 Technical Specifications require:

1. an exemption from the requirements of 10 CFR 50 Appendix G to permit application of ASME Section XI Code Case N-640 [6] to North Anna Units 1 and 2, and
2. an exemption from the requirements of 10 CFR 50 Appendix G to permit plant-specific application of the analysis methodology that supports ASME Section XI Code Case N-514 [8] to North Anna Units 1 and 2.

Justifications for the required exemptions are provided in Section 6.1 and 6.2, and Appendices E and F.

6.1 ASME Section XI Code Case N-640

ASME Section XI Code Case N-640 [6] supports use of the ASME Section XI Appendix A K_{1c} fracture toughness curve (Figure A-4200-1), instead of the ASME Section XI Appendix G K_{1a} curve (Figure G-2210-1) that was employed in the development of the existing P/T limits and LTOPS setpoints [3] [4]. Appendix E provides justification for an exemption request to permit application of ASME Section XI Code Case N-640 to North Anna Units 1 and 2.

6.2 Alternate LTOPS T_{enable} Methodology

The current North Anna Units 1 and 2 Technical Specification T_{enable} values are established at RT_{NDT} + 50°F + temperature measurement uncertainty [3] [4]. The T_{enable} design was based on recommendations of the ASME Section XI Working Group on Operating Plant Criteria. Subsequent to the Reference [3] submittal, the NRC has adopted Code Case N-514 into 10 CFR 50 Appendix G through 10 CFR 50.55a, "Codes and Standards". However, an exemption from the requirements of 10 CFR 50 Appendix G is required to permit a plant-specific application of the analysis methodology that supports ASME Section XI Code Case N-514 to North Anna Units 1 and 2. The proposed analysis methodology is presented in Appendix D, and justification for an exemption request is presented in Appendix F.

7.0 Safety Significance

Virginia Power proposes modification of the North Anna Units 1 and 2 Technical Specifications to extend the cumulative core burnup applicability limits for the Units 1 and 2 P/T limits, LTOPS setpoints, and T_{enable} values to 32.3 EFPY and 34.3 EFPY, respectively. Changes to the supporting analysis bases include:

1. Replacement of the current design and licensing basis P/T limits, including the isothermal (steady-state) P/T limit curve that constitutes the design limit for the LTOPS setpoint analysis, with those documented in Appendix C [7],
2. Replacement of the current design and licensing basis RT_{NDT} calculations, and the associated relationship of cumulative core burnup to reactor vessel neutron fluence, with those previously submitted in Reference [2],
3. Modification of the analysis basis for the Technical Specification LTOPS T_{enable} values with a plant-specific implementation of the analysis methodology that supports ASME Section XI Code Case N-514 [8].

Implementation of these proposed revised analysis bases requires:

1. An exemption from the requirements of 10 CFR 50 Appendix G to permit application of ASME Section XI Code Case N-640 [6] to North Anna Units 1 and 2, and
2. An exemption from the requirements of 10 CFR 50 Appendix G to permit plant-specific application of the analysis methodology that supports ASME Section XI Code Case N-514 [8] to North Anna Units 1 and 2.

The proposed revised analysis bases support continued use of the existing North Anna Units 1 and 2 Technical Specification P/T limit curves, LTOPS setpoints, LTOPS enable temperatures for North Anna Units 1 and 2 cumulative core burnups up to 32.3 EFPY and 34.3 EFPY, respectively. The supporting analyses demonstrate that established analysis acceptance criteria continue to be met. Specifically, the existing P/T limit curves, LTOPS setpoints, and LTOPS T_{enable} values provide acceptable margin to vessel fracture under both normal operation and LTOPS design basis (mass addition and heat addition) accident conditions. Thus, the margin of safety inherent in the P/T limits and LTOPS design analyses is not decreased by the proposed changes. No changes to plant systems, structures, or components are proposed, and no new allowable operating modes are established. Therefore, the proposed changes do not result in a new or different type of accident, since no new accident precursors are created. Because the proposed revised licensing basis analyses utilize acceptable analytical methods, and continue to demonstrate that established analysis acceptance criteria continue to be met, the consequences of accidents previously analyzed are not increased.

8.0 References

[1] BAW-2356, "Analysis of Capsule W, Virginia Power North Anna Unit No. 1 Nuclear Power Plant, Reactor Vessel Material Surveillance Program," dated September 1999.

[2] Letter from L. N. Hartz to USNRC, "Virginia Electric and Power Company, North Anna Power Station Units 1 and 2, Surry Power Station Units 1 and 2, Evaluation of Reactor Vessel Materials Surveillance Data," dated November 19, 1999 (Virginia Power Serial No. 99-452A).

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

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

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

[6] ASME Code Section XI, Code Case N-640, "Revision to Appendix G – Use of K_{Ic}," Approved by Section XI and Main Committee, Published in May 1999, and included in the 1999 Addenda.

[7] WCAP-15112, Revision 1, "North Anna Units 1 and 2 WOG Reactor Vessel 60-Year Evaluation Minigroup Heatup and Cooldown Limit Curves for Normal Operation," dated October, 1998.

[8] ASME Code Section XI, Code Case N-514, "Low Temperature Overpressure Protection."

[9] J. M. Chicots and M. J. Malone: "Heatup and Cooldown Curves for North Anna Unit 1," WCAP-13831, Revision 1, dated November 1993.

[10] N. K. Ray, et al.: "North Anna Unit 2, Reactor Vessel Heatup and Cooldown Limit Curves for Normal Operation (Capsule U)," WCAP-12503, dated March 1990.

APPENDIX A

North Anna Unit 1 P/T Limits
(From WCAP- 13831, Revision 1 [11])

Table 2	North Anna Unit 1 Cooldown Data with Margins of 0 Degrees F and 0 psi for Instrumentation Errors (WCAP-13831, Revision 1)

Cooldown Rate = 0 Deg. F/hr

	Indicated Temperature (Deg. F)	Indicated Pressure (psig)
1	85	547.28
2	90	552.65
3	95	558.42
4	100	564.63
5	105	571.30
6	110	578.47
7	115	586.06
8	120	594.35
9	125	603.26
10	130	612.85
11	135	623.15
12	140	634.08
13	145	645.99
14	150	658.79
15	155	672.54
16	160	687.19
17	165	703.09
18	170	720.16
19	175	738.39
20	180	758.16
21	185	779.19
22	190	802.03
23	195	826.34
24	200	852.71
25	205	880.84
26	210	911.22
27	215	943.79
28	220	978.72
29	225	1016.23
30	230	1056.54
31	235	1100.08
32	240	1146.64
33	245	1196.62
34	250	1250.15
35	255	1307.90
36	260	1369.68
37	265	1435.87
38	270	1507.19
39	275	1583.43
40	280	1664.94
41	285	1752.74
42	290	1846.59
43	295	1946.83
44	300	2054.44
45	305	2169.27
46	310	2292.07
47	315	2423.12

Cooldown Rate = 20 Deg. F/hr

	Indicated Temperature (Deg. F)	Indicated Pressure (psig)
1	85	513.64
2	90	519.15
3	95	525.11
4	100	531.51
5	105	538.32
6	110	545.75
7	115	553.77
8	120	562.38
9	125	571.68
10	130	581.66
11	135	592.32
12	140	603.90
13	145	616.38
14	150	629.77
15	155	644.11
16	160	659.65
17	165	676.35
18	170	694.21
19	175	713.59
20	180	734.23
21	185	756.66
22	190	780.55
23	195	806.48
24	200	834.16
25	205	864.11
26	210	896.19
27	215	930.62
28	220	967.62
29	225	1007.61
30	230	1050.44
31	235	1096.47
32	240	1145.88

Cooldown Rate = 40 Deg. F/hr

	Indicated Temperature (Deg. F)	Indicated Pressure (psig)
1	85	479.44
2	90	485.12
3	95	491.20
4	100	497.82
5	105	505.00
6	110	512.72
7	115	521.07
8	120	530.05
9	125	539.66
10	130	550.10
11	135	561.39
12	140	573.52
13	145	586.50
14	150	600.59
15	155	615.80
16	160	632.00
17	165	649.64
18	170	668.61
19	175	688.90
20	180	710.90
21	185	734.42
22	190	759.89
23	195	787.16
24	200	816.63
25	205	848.28
26	210	882.22
27	215	918.77
28	220	958.26
29	225	1000.60
30	230	1046.08
31	235	1094.99

Table 1	North Anna Unit 1 Heatup Data with Margins of 0 Degrees F and 0 psi for Instrumentation Errors (WCAP-13831, Revision 1)

Heatup Rate = 20 Deg. F/hr

	Indicated Temperature (Deg. F)	Indicated Pressure (psig)
1	85	541.05
2	90	544.65
3	95	551.19
4	100	558.02
5	105	566.12
6	110	574.67
7	115	583.99
8	120	594.05
9	125	603.26
10	130	612.85
11	135	623.15
12	140	634.08
13	145	645.99
14	150	658.79
15	155	672.54
16	160	687.19
17	165	703.09
18	170	720.16
19	175	738.39
20	180	758.16
21	185	779.19
22	190	802.03
23	195	826.34
24	200	852.71
25	205	880.84
26	210	911.22
27	215	943.79
28	220	978.72
29	225	1016.23
30	230	1056.54
31	235	1100.08
32	240	1146.64
33	245	1196.62
34	250	1247.04
35	255	1301.47
36	260	1359.75
37	265	1422.22
38	270	1489.11
39	275	1561.00
40	280	1637.87
41	285	1720.33
42	290	1808.46
43	295	1903.06
44	300	2004.25
45	305	2112.48
46	310	2227.78
47	315	2351.31
48	320	2482.97

Heatup Rate = 40 Deg. F/hr

	Indicated Temperature (Deg. F)	Indicated Pressure (psig)
1	85	523.08
2	90	523.08
3	95	523.91
4	100	526.71
5	105	531.26
6	110	536.96
7	115	543.97
8	120	551.92
9	125	560.88
10	130	570.68
11	135	581.44
12	140	592.98
13	145	605.64
14	150	619.27
15	155	633.87
16	160	649.71
17	165	666.81
18	170	685.00
19	175	704.79
20	180	725.84
21	185	748.72
22	190	773.22
23	195	799.50
24	200	827.64
25	205	858.12
26	210	890.69
27	215	925.64
28	220	963.17
29	225	1003.73
30	230	1047.10
31	235	1093.62
32	240	1143.61
33	245	1196.62
34	250	1246.41
35	255	1297.66
36	260	1352.70
37	265	1411.55
38	270	1474.75
39	275	1542.37
40	280	1614.86
41	285	1692.59
42	290	1775.78
43	295	1864.70
44	300	1960.06
45	305	2061.90
46	310	2170.71
47	315	2286.98
48	320	2410.81

Heatup Rate = 60 Deg. F/hr

	Indicated Temperature (Deg. F)	Indicated Pressure (psig)
1	85	507.18
2	90	507.18
3	95	507.18
4	100	507.18
5	105	507.82
6	110	510.04
7	115	513.81
8	120	518.78
9	125	525.00
10	130	532.25
11	135	540.50
12	140	549.81
13	145	560.20
14	150	571.56
15	155	583.91
16	160	597.43
17	165	612.13
18	170	628.00
19	175	645.05
20	180	663.57
21	185	683.38
22	190	704.87
23	195	727.84
24	200	752.74
25	205	779.33
26	210	808.11
27	215	838.87
28	220	871.89
29	225	907.59
30	230	945.78
31	235	986.80
32	240	1030.80
33	245	1078.06
34	250	1128.77
35	255	1183.17
36	260	1241.50
37	265	1304.05
38	270	1371.17
39	275	1443.05
40	280	1520.15
41	285	1602.65
42	290	1691.00
43	295	1785.48
44	300	1886.69
45	305	1994.52
46	310	2110.10
47	315	2229.35
48	320	2346.49
49	325	2471.30

Table 2 (Cont'd)	North Anna Unit 1 Cooldown Data with Margins of 0 Degrees F and 0 psi for Instrumentation Errors (WCAP-13831, Revision 1)

Cooldown Rate = 60 Deg. F/hr

	Indicated Temperature (Deg. F)	Indicated Pressure (psig)
1	85	444.59
2	90	450.47
3	95	456.85
4	100	463.73
5	105	471.20
6	110	479.24
7	115	487.96
8	120	497.26
9	125	507.42
10	130	518.37
11	135	530.21
12	140	542.85
13	145	556.63
14	150	571.46
15	155	587.36
16	160	604.60
17	165	623.22
18	170	643.11
19	175	664.74
20	180	687.84
21	185	712.94
22	190	739.76
23	195	768.82
24	200	799.99
25	205	833.52
26	210	869.58
27	215	908.64
28	220	950.46
29	225	995.48
30	230	1043.85
31	235	1095.93

Cooldown Rate = 100 Deg. F/hr

	Indicated Temperature (Deg. F)	Indicated Pressure (psig)
1	85	373.22
2	90	379.55
3	95	386.47
4	100	393.94
5	105	402.08
6	110	410.82
7	115	420.38
8	120	430.71
9	125	441.85
10	130	453.95
11	135	467.07
12	140	481.22
13	145	496.47
14	150	513.00
15	155	530.89
16	160	550.08
17	165	570.94
18	170	593.30
19	175	617.60
20	180	643.63
21	185	671.90
22	190	702.20
23	195	734.90
24	200	770.25
25	205	808.29
26	210	849.22
27	215	893.33
28	220	940.81
29	225	991.95
30	230	1046.95

Table 2 (Page 2 of 2) (From WCAP-13831 Rev. 1)

APPENDIX B

North Anna Unit 2 P/T Limits
(WCAP-12503)

Table 1	North Anna Unit 2 Heatup Data with Margins of 0 Degrees F and 0 psi for Instrumentation Errors (WCAP-12503)

Heatup Rate = 20 Deg. F/hr			Heatup Rate = 40 Deg. F/hr			Heatup Rate = 60 Deg. F/hr		
	Indicated Temperature (Deg. F)	Indicated Pressure (psig)		Indicated Temperature (Deg. F)	Indicated Pressure (psig)		Indicated Temperature (Deg. F)	Indicated Pressure (psig)
1	85	513.18	1	85	493.06	1	85	473.41
2	90	514.28	2	90	493.06	2	90	473.41
3	95	518.10	3	95	493.06	3	95	473.41
4	100	522.09	4	100	493.27	4	100	473.41
5	105	527.12	5	105	495.05	5	105	473.41
6	110	532.41	6	110	497.85	6	110	473.41
7	115	538.35	7	115	501.67	7	115	474.31
8	120	544.65	8	120	506.10	8	120	476.18
9	125	551.44	9	125	511.40	9	125	479.10
10	130	558.79	10	130	517.28	10	130	482.79
11	135	566.76	11	135	523.86	11	135	487.34
12	140	575.29	12	140	531.00	12	140	492.59
13	145	584.52	13	145	538.83	13	145	498.61
14	150	594.38	14	150	547.16	14	150	505.22
15	155	604.47	15	155	556.35	15	155	512.67
16	160	613.68	16	160	566.21	16	160	520.82
17	165	623.59	17	165	576.89	17	165	529.76
18	170	634.24	18	170	588.36	18	170	539.46
19	175	645.54	19	175	600.63	19	175	549.90
20	180	657.85	20	180	613.92	20	180	561.29
21	185	671.08	21	185	628.27	21	185	573.63
22	190	685.29	22	190	643.65	22	190	586.93
23	195	700.43	23	195	660.12	23	195	601.16
24	200	716.88	24	200	677.95	24	200	616.62
25	205	734.54	25	205	696.97	25	205	633.29
26	210	753.36	26	210	717.58	26	210	651.06
27	215	773.79	27	215	739.72	27	215	670.37
28	220	795.53	28	220	763.38	28	220	691.09
29	225	819.14	29	225	788.95	29	225	713.28
30	230	844.27	30	230	816.32	30	230	737.25
31	235	871.53	31	235	845.67	31	235	762.91
32	240	900.61	32	240	877.42	32	240	790.59
33	245	932.01	33	245	911.38	33	245	820.26
34	250	965.68	34	250	947.79	34	250	852.06
35	255	1001.79	35	255	986.92	35	255	886.40
36	260	1040.57	36	260	1029.17	36	260	923.17
37	265	1082.23	37	265	1074.33	37	265	962.68
38	270	1127.24	38	270	1122.83	38	270	1005.04
39	275	1175.32	39	275	1174.89	39	275	1050.55
40	280	1226.73	40	280	1226.82	40	280	1099.40
41	285	1278.89	41	285	1278.19	41	285	1151.82
42	290	1334.84	42	290	1330.79	42	290	1208.04
43	295	1394.87	43	295	1387.15	43	295	1268.40
44	300	1459.10	44	300	1447.61	44	300	1332.98
45	305	1528.12	45	305	1512.06	45	305	1402.35
46	310	1601.98	46	310	1581.69	46	310	1476.62
47	315	1681.18	47	315	1655.78	47	315	1556.29
48	320	1766.00	48	320	1735.73	48	320	1641.53
49	325	1856.92	49	325	1820.99	49	325	1732.71
50	330	1954.10	50	330	1912.20	50	330	1830.42
51	335	2058.33	51	335	2009.92	51	335	1934.85
52	340	2169.49	52	340	2114.39	52	340	2046.38
53	345	2288.42	53	345	2225.88	53	345	2165.60
54	350	2415.28	54	350	2345.12	54	350	2282.71
			55	355	2472.24	55	355	2402.28

Table 2	North Anna Unit 2 Cooldown Data with Margins of 0 Degrees F and 0 psi for Instrumentation Errors (WCAP-12503)

Cooldown Rate = 0 Deg. F/hr

	Indicated Temperature (Deg. F)	Indicated Pressure (psig)
1	85	526.54
2	90	529.88
3	95	533.48
4	100	537.34
5	105	541.49
6	110	545.96
7	115	550.65
8	120	555.81
9	125	561.36
10	130	567.33
11	135	573.74
12	140	580.64
13	145	588.05
14	150	595.90
15	155	604.47
16	160	613.68
17	165	623.59
18	170	634.24
19	175	645.54
20	180	657.85
21	185	671.08
22	190	685.29
23	195	700.43
24	200	716.88
25	205	734.54
26	210	753.36
27	215	773.79
28	220	795.53
29	225	819.14
30	230	844.27
31	235	871.53
32	240	900.61
33	245	932.01
34	250	965.68
35	255	1001.79
36	260	1040.57
37	265	1082.23
38	270	1127.24
39	275	1175.32
40	280	1226.82
41	285	1282.38
42	290	1342.01
43	295	1405.87
44	300	1474.37
45	305	1547.92
46	310	1626.80
47	315	1711.17
48	320	1801.58
49	325	1898.56
50	330	2002.43
51	335	2113.50
52	340	2231.98
53	345	2358.87

Cooldown Rate = 20 Deg. F/hr

	Indicated Temperature (Deg. F)	Indicated Pressure (psig)
1	85	488.96
2	90	492.32
3	95	495.97
4	100	499.89
5	105	504.04
6	110	508.60
7	115	513.54
8	120	518.84
9	125	524.57
10	130	530.72
11	135	537.38
12	140	544.53
13	145	552.14
14	150	560.43
15	155	569.39
16	160	579.01
17	165	589.39
18	170	600.41
19	175	612.44
20	180	625.37
21	185	639.29
22	190	654.13
23	195	670.26
24	200	687.59
25	205	706.12
26	210	726.21
27	215	747.64
28	220	770.89
29	225	795.70
30	230	822.61
31	235	851.33
32	240	882.39
33	245	915.70
34	250	951.41
35	255	989.83
36	260	1031.31
37	265	1075.78
38	270	1123.49
39	275	1174.83

Cooldown Rate = 40 Deg. F/hr

	Indicated Temperature (Deg. F)	Indicated Pressure (psig)
1	85	450.56
2	90	453.96
3	95	457.67
4	100	461.67
5	105	466.01
6	110	470.69
7	115	475.77
8	120	481.23
9	125	487.16
10	130	493.54
11	135	500.46
12	140	507.80
13	145	515.86
14	150	524.52
15	155	533.89
16	160	543.96
17	165	554.75
18	170	566.46
19	175	579.12
20	180	592.72
21	185	607.30
22	190	623.10
23	195	640.15
24	200	658.35
25	205	678.13
26	210	699.23
27	215	722.17
28	220	746.64
29	225	773.22
30	230	801.59
31	235	832.35
32	240	865.26
33	245	900.66
34	250	938.71
35	255	979.89
36	260	1023.98
37	265	1071.42
38	270	1122.35

Table 2 (Cont'd)	North Anna Unit 2 Cooldown Data with Margins of 0 Degrees F and 0 psi for Instrumentation Errors (WCAP-12503)

Cooldown Rate = 60 Deg. F/hr

	Indicated Temperature (Deg. F)	Indicated Pressure (psig)
1	85	411.52
2	90	414.92
3	95	418.70
4	100	422.79
5	105	427.25
6	110	432.06
7	115	437.31
8	120	442.96
9	125	449.04
10	130	455.67
11	135	462.88
12	140	470.64
13	145	479.06
14	150	488.13
15	155	497.96
16	160	508.45
17	165	519.91
18	170	532.25
19	175	545.60
20	180	559.86
21	185	575.39
22	190	592.10
23	195	610.03
24	200	629.47
25	205	650.30
26	210	672.88
27	215	697.07
28	220	723.29
29	225	751.38
30	230	781.78
31	235	814.39
32	240	849.44
33	245	887.38
34	250	928.04
35	255	971.85
36	260	1018.93
37	265	1069.60
38	270	1124.03

Cooldown Rate = 100 Deg. F/hr

	Indicated Temperature (Deg. F)	Indicated Pressure (psig)
1	85	330.73
2	90	334.33
3	95	338.31
4	100	342.63
5	105	347.38
6	110	352.53
7	115	358.18
8	120	364.29
9	125	370.94
10	130	378.17
11	135	386.06
12	140	394.59
13	145	403.87
14	150	413.84
15	155	424.75
16	160	436.52
17	165	449.22
18	170	463.01
19	175	477.96
20	180	494.08
21	185	511.46
22	190	530.29
23	195	550.56
24	200	572.52
25	205	596.15
26	210	621.75
27	215	649.26
28	220	679.06
29	225	711.08
30	230	745.55
31	235	782.93
32	240	823.05
33	245	866.28
34	250	912.81
35	255	962.99
36	260	1016.92
37	265	1075.03

Table 2 (Page 2 of 2) (From WCAP-12503)

APPENDIX C

Pressure/Temperature Limits Development for North Anna Units 1 and 2

Background

Revised North Anna Units 1 and 2 reactor coolant system (RCS) pressure-temperature operating limit curves (i.e., heatup and cooldown limit curves) for normal operation were developed in WCAP-15112, Revision 1 [C-1]. The curves were developed to support operation during a postulated 20-year license renewal period for North Anna Units 1 and 2. The portions WCAP-15112, Revision 1 [C-1] that are applicable to the present analysis have been excerpted, and are presented below. Certain minor editorial modifications to the Reference [C-1] text were necessary, since only the heatup and cooldown limit curve analysis performed using the ASME Section XI Appendix G K_{1c} fracture toughness methodology is presented herein.

Criteria for Allowable Pressure-Temperature Relationships

Appendix G to 10 CFR 50, "Fracture Toughness Requirements" [C-5] specifies fracture toughness requirements for ferritic materials of pressure-retaining components of the reactor coolant pressure boundary of light water power reactors to provide adequate margins of safety during any condition of normal operation, including anticipated operational occurrences and system hydrostatic tests, to which the pressure boundary may be subjected over its service lifetime. The ASME Boiler and Pressure Vessel Code forms the basis for these requirements. Section XI, Division 1, "Rules for Inservice Inspection of Nuclear Power Plant Components," Appendix G [C-2], contains the conservative methods of analysis.

The ASME approach for calculating the allowable limit curves for various heatup and cooldown rates specifies that the total stresses at any time during heatup or cooldown cannot be greater than the reference stress intensity factor, K_{1c}, for the metal temperature at that time. K_{1c} is obtained from the reference fracture toughness curve and is given by the following equation:

$$K_{1c} = 33.2 + 20.734e^{[0.02\,(T-RT_{NDT})]} \quad (1)$$

where K_{1c} is the reference stress intensity factor as a function of metal temperature T and the metal reference nil-ductility temperature RT_{NDT}.

Therefore, the governing equation for the heatup and cooldown curve analysis is defined as follows:

$$C * K_{1m} + K_{1t} < K_{1c} \quad (2)$$

where

K_{1m} = the stress intensity factor caused by membrane (pressure) stress
K_{1t} = the stress intensity factor caused by the thermal gradients
K_{1c} = a function of temperature relative to the RT_{NDT} of the material
C = 2.0 for Level A and B service limits
C = 1.5 for hydrostatic and leak test conditions during which the reactor core is not critical

At any time during the heatup or cooldown transient, K_{1c} is determined by the metal temperature at the tip of a postulated flaw at the ¼-T and ¾-T, the appropriate value for RT_{NDT}, and the reference fracture toughness curve. The thermal stresses resulting from the temperature gradients through the vessel wall are calculated and then the corresponding (thermal) stress intensity factors, K_{1t}, for the reference flaw are then computed. From Equation 2, the pressure stress intensity factors are obtained and, from these, the allowable pressures are calculated.

For the calculation of the allowable pressure versus coolant temperature during cooldown, the reference flaw of Appendix G to the ASME Code is assumed to exist at the inside of the vessel wall. During cooldown, the controlling location of the flaw is always at the inside of the wall because the thermal gradients produce tensile stresses at the inside, which increase with increasing cooldown rates. Allowable pressure-temperature relations are generated for both steady-state and finite cooldown rates situations. From these relations, composite limit curves are constructed for each cooldown rate of interest.

The use of the composite curve in the cooldown analysis is necessary because control of the cooldown procedure is based on the measurement of reactor coolant temperature, whereas the limiting pressure is actually dependent on the material temperature at the tip of the assumed flaw. During cooldown, the ¼-T vessel location is at a higher temperature than the fluid adjacent to the vessel inner diameter. This condition, of course, is not true for the steady-state situation. It follows that, at any given reactor coolant temperature, the ΔT (differential temperature) developed during cooldown results in a higher value K_{1c} at the ¼-T location for finite cooldown rates than for steady-state operation. Furthermore, if conditions exist so that the increase in K_{1c} exceeds K_{1t}, the calculated allowable pressure during cooldown is greater than the steady-state value.

The above procedures are needed because there is no direct control on temperature at the ¼-T location and, therefore, allowable pressures may unknowingly be violated if the rate of cooling is decreased at various intervals along the cooldown ramp. The use of the composite curve eliminates this problem and ensures conservative operation of the system for the entire cooldown period.

Three separate calculations are required to determine the limit curves for finite heatup rates. As is done in the cooldown analysis, allowable pressure-temperature relationships are developed for steady-state conditions as well as finite heatup rate conditions assuming the presence of a ¼-T defect at the inside of the wall. The heatup results in compressive stresses at the inside surface that alleviate the tensile stresses produced by internal pressure. The metal temperature at the crack tip lags the coolant temperature; therefore, the K_{1c} for the ¼-T crack during heatup is lower than the K_{1c} for the ¼-T crack during steady-state conditions at the same coolant temperature. During heatup, especially at the end of the transient, conditions may exist so that the effects of compressive thermal stresses and lower K_{1c} values do not offset each other, and the pressure-temperature curve based on steady-state conditions no longer represents a lower bound of all similar curves for finite heatup rates when the ¼-T flaw is considered. Therefore, both cases have to be analyzed in order to ensure that at any coolant temperature the lower value of the allowable

The second portion of the heatup analysis concerns the calculation of the pressure-temperature limitations for the case in which a ¼-T flaw located at the ¼-T location from the outside surface is assumed. Unlike the situation at the vessel inside surface, the thermal gradients established at the outside surface during heatup produce stresses which are tensile in nature and therefore tend to reinforce any pressure stresses present. These thermal stresses are dependent on both the rate of heatup and the time (or coolant temperature) along the heatup ramp. Since the thermal stresses at the outside are tensile and increase with increasing heatup rates, each heatup rate must be analyzed on an individual basis.

Following the generation of pressure-temperature limit curve for both the steady-state and finite heatup rate situations, the final limit curves are produced by constructing a composite curve based on a point-by-point comparison of the steady-state and finite heatup rate data. At any given temperature, the allowable pressure is taken to be the lesser of the three values taken from the curves under consideration. The use of the composite curve is necessary to set conservative heatup limitations because it is possible for conditions to exist wherein, over the course of the heatup ramp, the controlling conditions switches from the inside to the outside, and the pressure limit must at all times be based on analysis of the most critical criterion.

10 CFR 50, Appendix G [C-5] addresses the metal temperature of the closure head flange and vessel flange regions. This rule states that the metal temperature of the closure flange regions must exceed the material unirradiated RT_{NDT} by at least 120°F for normal operation when the pressure exceeds 20 percent of the preservice hydrostatic test pressure (3107 psig), which is 621 psig [C-3] for the North Anna Units 1 and 2 reactor vessels.

The limiting unirradiated RT_{NDT} of -22°F occurs in the vessel flange of the North Anna Units 1 and 2 reactor vessels, so the minimum allowable temperature of this region is 98°F at pressures greater than 621 psig with uncertainties of 0°F and 0 psi. This limit is reflected in the heatup and cooldown curves shown in Figures 1 through 4.

Reactor Vessel Geometric & System Parameters

The applicable reactor vessel physical dimensions and operating conditions, along with other system parameters, are shown in the following table:

Reactor Vessel Physical Dimensions and Operating Conditions (From Table 6-1 of WCAP-11512 Revision 1 [C-1])	
Parameter	Value
Vessel Beltline Thickness	7.705 inches
Vessel Inner Radius to Clad	78.95 inches
Vessel Clad Thickness	0.16 inches
Pre-Service Hydrostatic Pressure	3107 psig
System and Component Operating Conditions/Dimensions	Design Pressure = 2485 psig
	Operating Pressure = 2235 psig

The reactor vessel may be bolted up at the initial RT_{NDT} of the material stressed by the boltup. The most limiting initial RT_{NDT} value is -22°F on the vessel flange. However, a minimum RCS termperature limit of 60°F is imposed to ensure that the RCS temperatures are sufficiently high to prevent damage to the closure head/vessel flange during the removal or installation of the reactor vessel head bolts.

Heatup and Cooldown Pressure-Temperature Limit Curves

Pressure-temperature limit curves for normal heatup and cooldown of the primary reactor coolant system have been calculated for the pressure and temperature in the reactor vessel beltline region using the methods described above. Figures 1 to 3 present the heatup curves with heatup rates of 20°F/hr, 40°F/hr, and 60°F/hr. (A heatup rate of 0°F/hr is defined by the steady-state cooldown curve). The curves include no margins for possible pressure or temperature instrumentation errors. Figure 4 presents the cooldown curves with cooldown rates of 0°F/hr, 20°F/hr, 40°F/hr, 60°F/hr, and 100°F/hr. Again, the curves include no margins for possible pressure and temperature instrumentation errors. The data points generated for developing the heatup and cooldown pressure-temperature limit curves are shown in Tables 1 and 2.

The reactor must not be made critical until pressure-temperature combinations are to the right of the criticality limit line shown in Figures 1 to 3 for the specific heatup rate and licensing period being utilized. The straight-line portion of the criticality limit is at the minimum permissible temperature for the 2485 psig inservice hydrostatic test as required by Appendix G to 10 CFR 50 [C-5]. The governing equations for the hydrostatic test is defined in Appendix G to Section XI of the ASME Code [C-2] as follows:

$$1.5\, K_{1m} < K_{1c}$$

where

K_{1m} is the stress intensity factor covered by membrane (pressure) stress
$K_{1c} = 33.2 + 20.734e^{[0.02\,(T-RT_{NDT})]}$
T is the minimum permissible metal temperature, and
RT_{NDT} is the metal reference nil-ductility temperature

The criticality limit specifies the pressure-temperature limits for core operation to provide additional margin during actual power production as specified in Reference [C-4]. The pressure-temperature limits for core operation (except for low power physics tests) are that the reactor vessel must be at a temperature equal to or higher than the minimum temperature required for the inservice hydrostatic test, and at least 40°F higher than the minimum permissible temperature in the corresponding pressure-temperature curve for heatup and cooldown calculated as described above. The vertical line drawn from these points on the pressure-temperature curve, intersecting a curve 40°F higher than the pressure-temperature limit curve, constitutes the limit for core operation for the reactor vessel.

References

[C-1] WCAP-15112, Revision 1, "North Anna Units 1 and 2 WOG Reactor Vessel 60-Year Evaluation Minigroup Heatup and Cooldown Limit Curves for Normal Operation," dated October, 1998.

[C-2] 1989 ASME Boiler and Pressure Vessel (B&PV) Code, Section XI, Appendix G, "Fracture Toughness Criteria for Protection Against Failure."

[C-3] 1989 Section III, Division 1 of the ASME Boiler and Pressure Vessel Code, Paragraph NB-2331, "Material for Vessels".

[C-4] 10 CFR 50.61, "Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events," Federal Register, Volume 60, No. 243, dated December 19, 1995.

[C-5] 10 CFR 50 Appendix G, "Fracture Toughness Requirements."

MATERIAL PROPERTY BASIS

LIMITING MATERIAL: Lower Shell Forging
LIMITING ART VALUES AT EOLR: 1/4T, 218.5°F
3/4T, 195.6°F

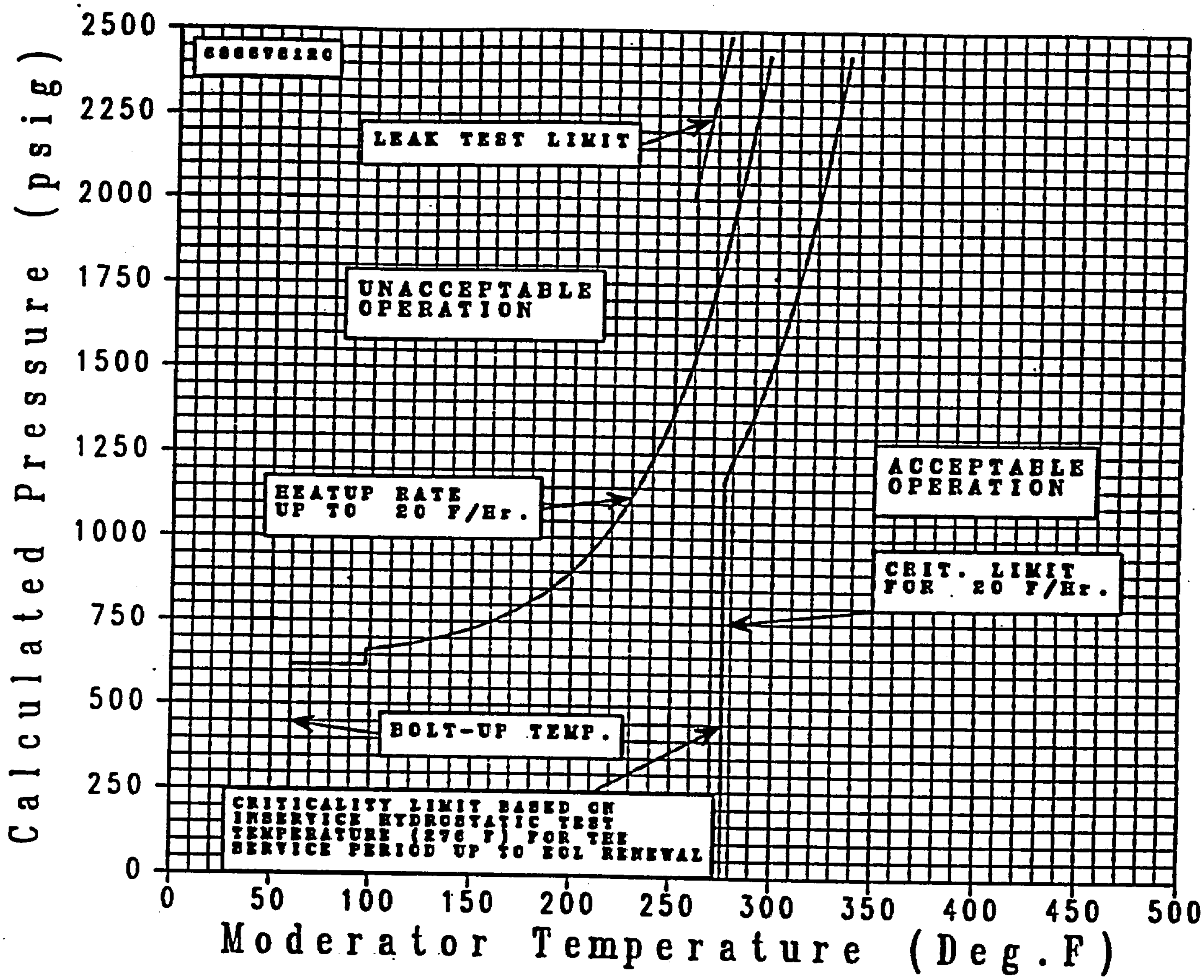

Figure ~~9-5~~ 1: North Anna Units 1 and 2 Reactor Coolant System Heatup Limitations (Heatup Rates of 20°F/hr) Applicable to End of License Renewal (With Margins of 0°F and 0 psi for Instrumentation Errors)

MATERIAL PROPERTY BASIS

LIMITING MATERIAL: Lower Shell Forging
LIMITING ART VALUES AT EOLR: 1/4T, 218.5°F
3/4T, 195.6°F

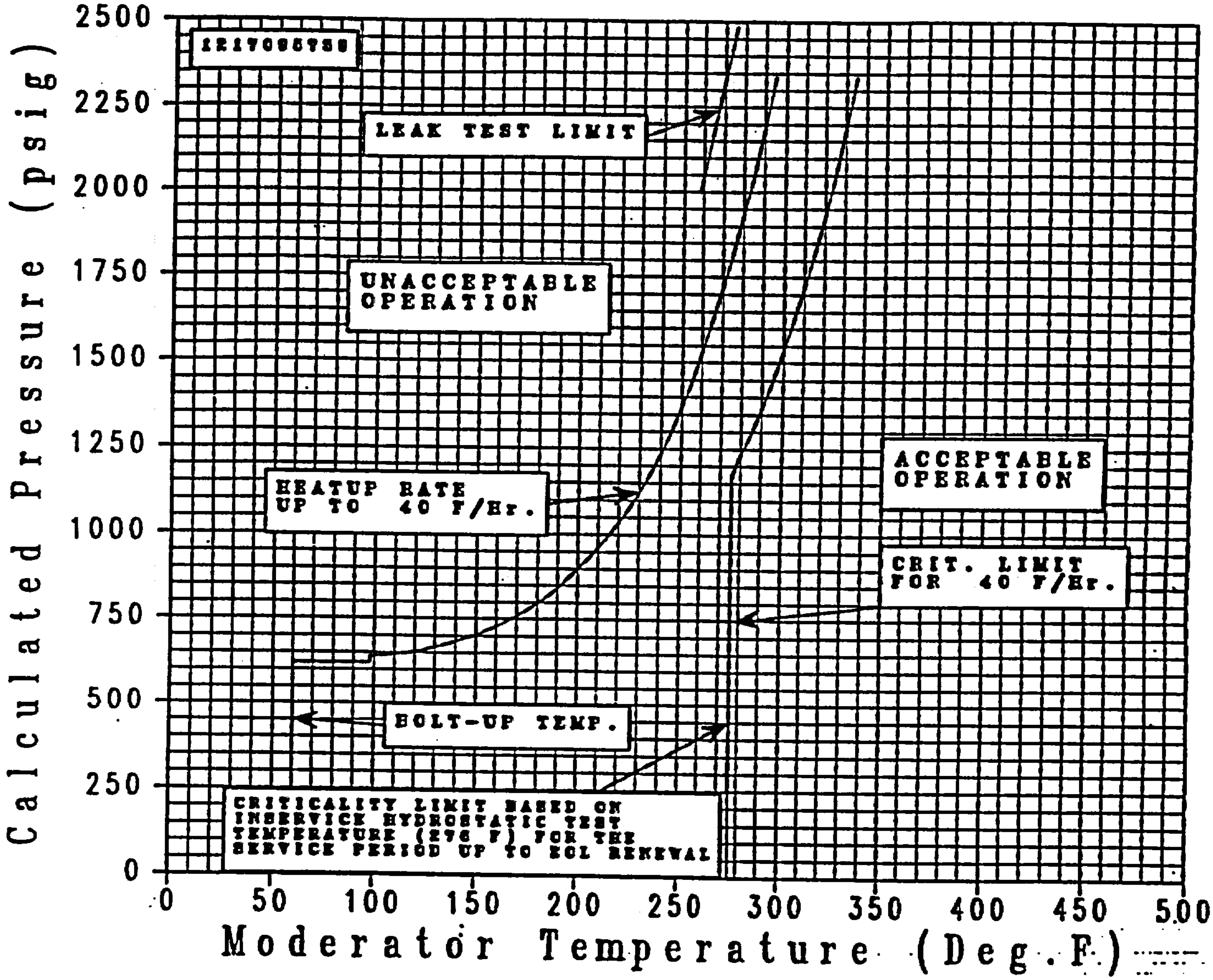

Figure ~~9-6~~ 2: North Anna Units 1 and 2 Reactor Coolant System Heatup Limitations (Heatup Rates of 40°F/hr) Applicable to End of License Renewal (With Margins of 0°F and 0 psi for Instrumentation Errors)

MATERIAL PROPERTY BASIS

LIMITING MATERIAL: Lower Shell Forging
LIMITING ART VALUES AT EOLR: 1/4T, 218.5°F
3/4T, 195.6°F

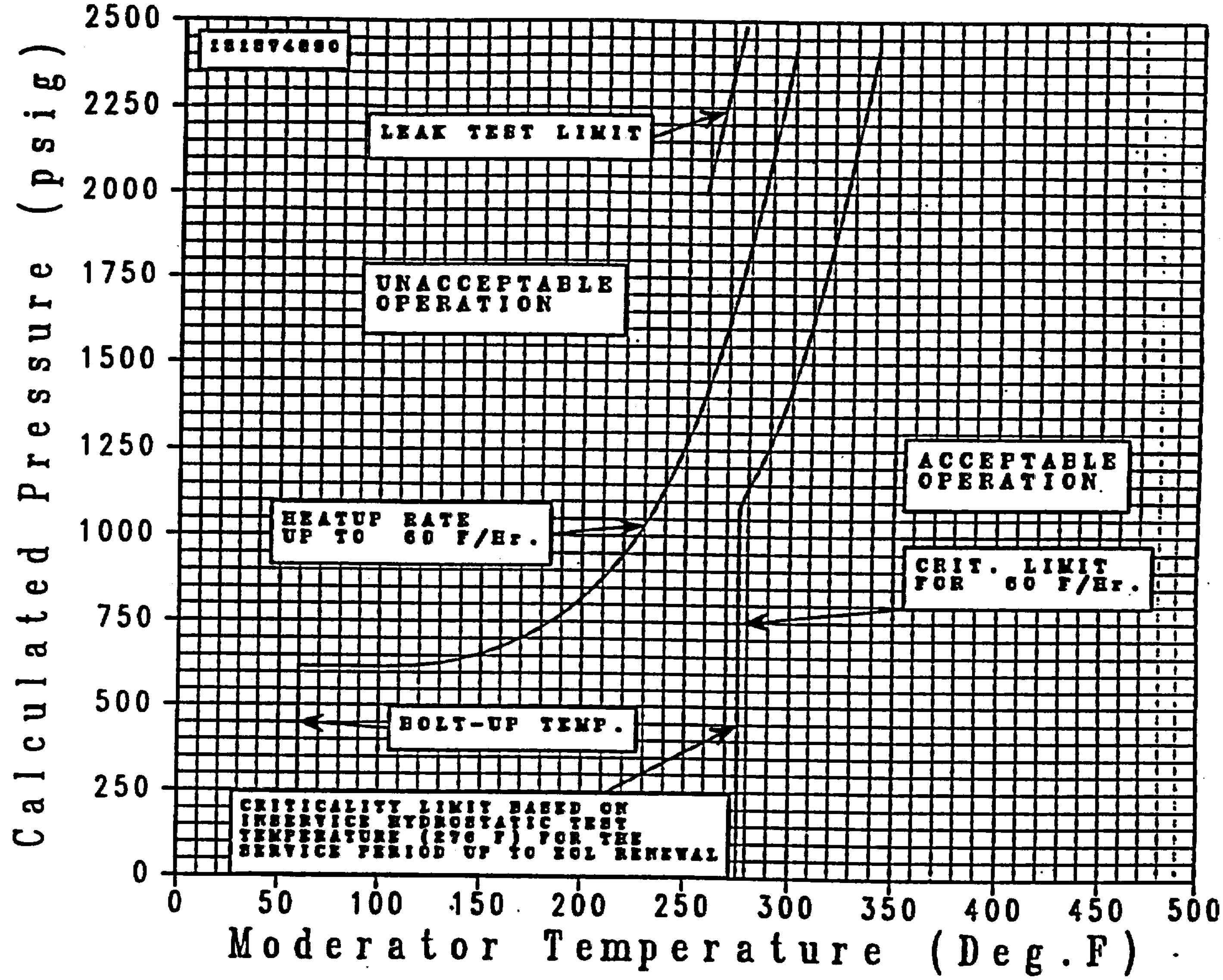

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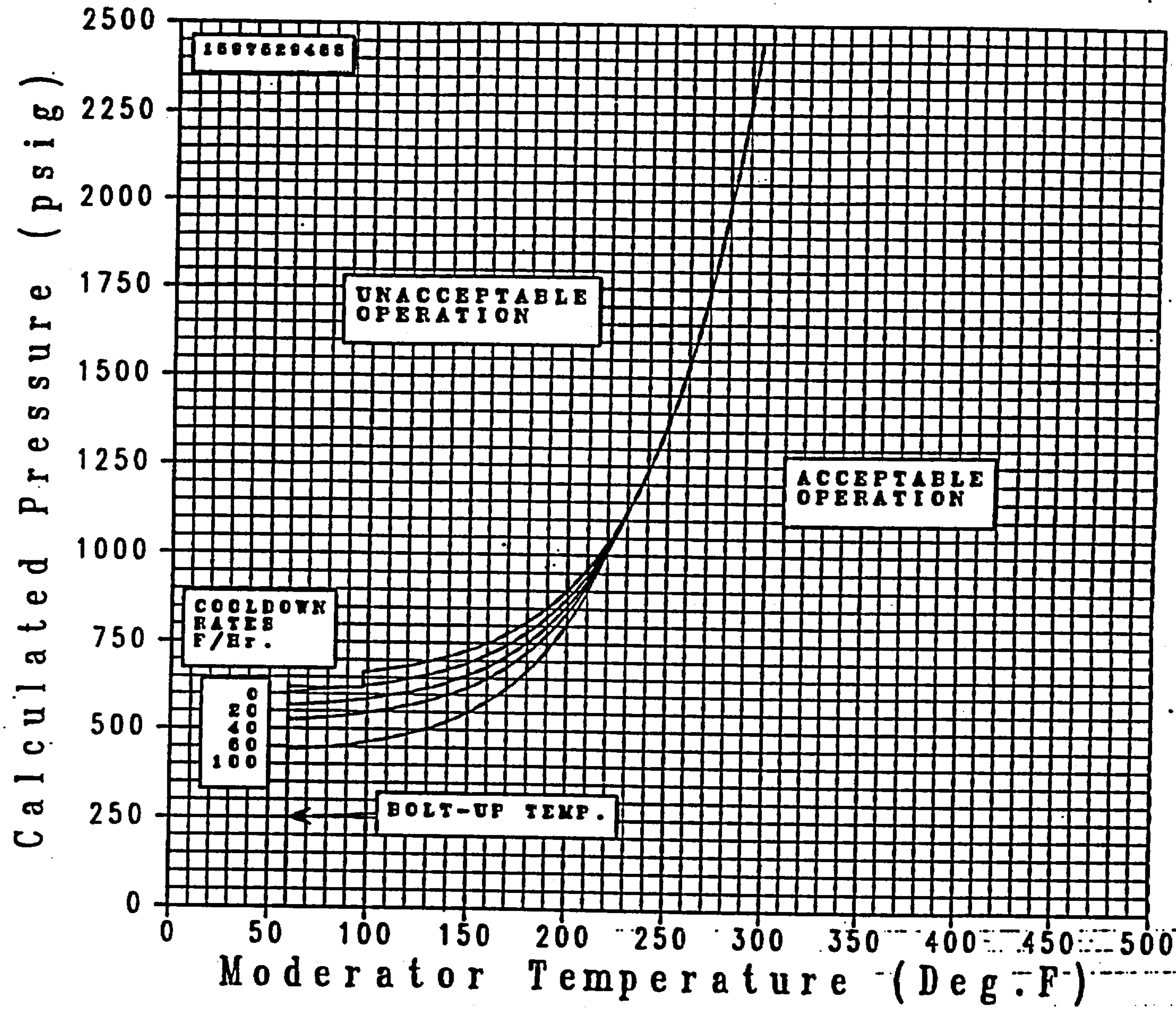
Figure ~~9-7~~: North Anna Units 1 and 2 Reactor Coolant System Heatup Limitations (Heatup Rates of 60°F/hr) Applicable to End of License Renewal (With Margins of 0°F and 0 psi for Instrumentation Errors)

MATERIAL PROPERTY BASIS

LIMITING MATERIAL: Lower Shell Forging
LIMITING ART VALUES AT EOLR: 1/4T, 218.5°F
3/4T, 195.6°F



4
Figure ~~9-8~~: North Anna Units 1 and 2 Reactor Coolant System Cooldown Limitations (Cooldown Rates of 0, 20, 40, 60 and 100 °F/hr) Applicable to End of License Renewal (With Margins of 0°F and 0 psi for Instrumentation Errors)

Table 1	North Anna Units 1 and 2 Heatup Data with Margins of 0 Degrees F and 0 psi for Instrumentation Errors (WCAP-15112 Rev. 1)

Heatup Rate = 20 Deg. F/hr

	Indicated Temperature (Deg. F)	Indicated Pressure (psig)
1	60	621.00
2	65	621.00
3	85	621.00
4	90	621.00
5	95	621.00
6	98	621.00
7	98	664.43
8	100	665.83
9	105	669.69
10	110	673.96
11	115	678.68
12	120	683.89
13	125	689.65
14	130	696.02
15	135	703.06
16	140	710.84
17	145	719.44
18	150	728.94
19	155	739.44
20	160	751.05
21	165	763.88
22	170	778.05
23	175	793.72
24	180	811.03
25	185	830.17
26	190	851.31
27	195	874.68
28	200	900.51
29	205	929.06
30	210	960.61
31	215	995.47
32	220	1034.00
33	225	1076.59
34	230	1123.65
35	235	1175.67
36	240	1233.15
37	245	1296.68
38	250	1366.89
39	255	1444.48
40	260	1530.24
41	265	1625.01
42	270	1729.76
43	275	1845.51
44	280	1973.45
45	285	2114.83
46	290	2265.93
47	295	2430.07

Heatup Rate = 40 Deg. F/hr

	Indicated Temperature (Deg. F)	Indicated Pressure (psig)
1	60	621.00
2	65	621.00
3	85	621.00
4	90	621.00
5	95	621.00
6	98	621.00
7	98	640.24
8	100	640.24
9	105	641.91
10	110	644.78
11	115	648.80
12	120	653.74
13	125	659.64
14	130	666.39
15	135	674.09
16	140	682.72
17	145	692.38
18	150	703.12
19	155	715.07
20	160	728.30
21	165	742.98
22	170	759.22
23	175	777.19
24	180	797.06
25	185	819.05
26	190	843.34
27	195	870.19
28	200	899.86
29	205	929.06
30	210	960.61
31	215	995.47
32	220	1034.00
33	225	1076.59
34	230	1123.65
35	235	1175.67
36	240	1233.15
37	245	1296.68
38	250	1366.89
39	255	1444.48
40	260	1530.24
41	265	1624.33
42	270	1716.12
43	275	1817.42
44	280	1929.31
45	285	2052.82
46	290	2189.25
47	295	2339.85

Heatup Rate = 60 Deg. F/hr

	Indicated Temperature (Deg. F)	Indicated Pressure (psig)
1	60	618.37
2	65	618.37
3	85	618.37
4	90	618.37
5	95	618.37
6	100	618.37
7	105	618.37
8	110	618.37
9	115	619.17
10	120	621.18
11	125	624.35
12	130	628.56
13	135	633.81
14	140	640.06
15	145	647.35
16	150	655.69
17	155	665.16
18	160	675.82
19	165	687.76
20	170	701.08
21	175	715.92
22	180	732.39
23	185	750.67
24	190	770.92
25	195	793.35
26	200	818.16
27	205	845.61
28	210	875.96
29	215	909.51
30	220	946.57
31	225	987.53
32	230	1032.76
33	235	1082.72
34	240	1137.90
35	245	1198.82
36	250	1266.08
37	255	1340.35
38	260	1422.32
39	265	1512.82
40	270	1612.71
41	275	1722.97
42	280	1844.66
43	285	1978.96
44	290	2123.44
45	295	2261.89
46	300	2414.77

Table 2	North Anna Units 1 and 2 Cooldown Data with Margins of 0 Degrees F and 0 psi for Instrumentation Errors (WCAP-15112 Rev. 1)

Cooldown Rate = 0 Deg. F/hr

	Indicated Temperature (Deg. F)	Indicated Pressure (psig)
1	60	621.00
2	65	621.00
3	70	621.00
4	75	621.00
5	80	621.00
6	85	621.00
7	90	621.00
8	95	621.00
9	98	621.00
10	98	664.43
11	100	665.83
12	105	669.69
13	110	673.96
14	115	678.68
15	120	683.89
16	125	689.65
17	130	696.02
18	135	703.06
19	140	710.84
20	145	719.44
21	150	728.94
22	155	739.44
23	160	751.05
24	165	763.88
25	170	778.05
26	175	793.72
27	180	811.03
28	185	830.17
29	190	851.31
30	195	874.68
31	200	900.51
32	205	929.06
33	210	960.61
34	215	995.47
35	220	1034.00
36	225	1076.59
37	230	1123.65
38	235	1175.67
39	240	1233.15
40	245	1296.68
41	250	1366.89
42	255	1444.48
43	260	1530.24
44	265	1625.01
45	270	1729.76
46	275	1845.51
47	280	1973.45
48	285	2114.83
49	290	2271.09
50	295	2443.78

Cooldown Rate = 20 Deg. F/hr

	Indicated Temperature (Deg. F)	Indicated Pressure (psig)
1	60	606.17
2	65	607.90
3	70	609.79
4	75	611.91
5	80	614.26
6	85	616.88
7	90	619.79
8	95	621.00
9	98	621.00
10	98	625.20
11	100	626.64
12	105	630.65
13	110	635.08
14	115	640.02
15	120	645.48
16	125	651.55
17	130	658.26
18	135	665.72
19	140	673.97
20	145	683.13
21	150	693.25
22	155	704.47
23	160	716.88
24	165	730.64
25	170	745.85
26	175	726.70
27	180	781.33
28	185	801.97
29	190	824.77
30	195	850.03
31	200	877.95
32	205	908.85
33	210	943.02
34	215	980.82
35	220	1022.61
36	225	1068.86
37	230	1119.98

Cooldown Rate = 40 Deg. F/hr

	Indicated Temperature (Deg. F)	Indicated Pressure (psig)
1	60	566.07
2	65	567.72
3	70	569.60
4	75	571.72
5	80	574.09
6	85	576.76
7	90	579.75
8	95	583.09
9	100	586.82
10	105	590.99
11	110	595.63
12	115	600.81
13	120	606.57
14	125	612.99
15	130	620.11
16	135	628.03
17	140	636.82
18	145	646.59
19	150	657.42
20	155	669.45
21	160	682.77
22	165	697.56
23	170	713.94
24	175	732.11
25	180	752.22
26	185	774.52
27	190	799.19
28	195	826.54
29	200	856.80
30	205	890.32
31	210	927.41
32	215	968.48
33	220	1013.91
34	225	1064.22
35	230	1119.86

Table 2 (Page 1 of 2) (From WCAP-15112 Rev. 1)

Table 2 (Cont'd)	North Anna Units 1 and 2 Cooldown Data with Margins of 0 Degrees F and 0 psi for Instrumentation Errors (WCAP-15112 Rev. 1)

Cooldown Rate = 60 Deg. F/hr

	Indicated Temperature (Deg. F)	Indicated Pressure (psig)
1	60	525.14
2	65	526.77
3	70	528.64
4	75	530.78
5	80	533.20
6	85	535.93
7	90	539.01
8	95	542.48
9	100	546.37
10	105	550.73
11	110	555.61
12	115	561.08
13	120	567.17
14	125	573.98
15	130	581.56
16	135	590.02
17	140	599.42
18	145	609.89
19	150	621.52
20	155	634.45
21	160	648.80
22	165	664.75
23	170	682.44
24	175	702.08
25	180	723.85
26	185	748.00
27	190	774.77
28	195	804.44
29	200	837.32
30	205	873.76
31	210	914.11
32	215	958.82
33	220	1008.33
34	225	1063.16

Cooldown Rate = 100 Deg. F/hr

	Indicated Temperature (Deg. F)	Indicated Pressure (psig)
1	60	440.84
2	65	442.46
3	70	444.38
4	75	446.60
5	80	449.15
6	85	452.08
7	90	455.42
8	95	459.22
9	100	463.52
10	105	468.39
11	110	473.88
12	115	480.05
13	120	486.98
14	125	494.76
15	130	503.47
16	135	513.21
17	140	524.09
18	145	536.24
19	150	549.78
20	155	564.89
21	160	581.70
22	165	600.41
23	170	621.22
24	175	644.37
25	180	670.08
26	185	698.65
27	190	730.36
28	195	765.58
29	200	804.65
30	205	848.01
31	210	896.10
32	215	949.44
33	220	1008.56
34	225	1074.11

Table 2 (Page 2 of 2) (From WCAP-15112 Rev. 1)

APPENDIX D

Revised LTOPS Enable Temperature Basis For North Anna Units 1 and 2

Acknowledgements

Virginia Power acknowledges the contributions of James R. Pfefferle of Wisconsin Electric Power Company and Thomas D. Spry of Guardian Metallurgy, Inc. toward the development the alternate Low Temperature Overpressure Protection System (LTOPS) enable temperature (T_{enable}) methodology described herein. Their draft ASME Section XI Code Case basis document entitled "Basis for Plant-Specific Low Temperature Ovepressure Protection System Enable Temperature" was used extensively by Virginia Power to establish a technical justification for the revised North Anna Units 1 and 2 plant-specific T_{enable} values documented herein. In their draft basis document, Mr. Pfefferle and Mr. Spry acknowledged the contributions of the ASME Section XI Working Group on Operating Plant Criteria. The suggestions of Mr. W. Bamford, Mr. R. Gamble, Mr. T. Griesbach, Mr. K. Wichman, and Dr. K. Yoon were noted as being particularly beneficial in completing their work.

Background

As older pressurized water reactors (PWRs) with high copper welds approach the end of their operating licenses and make the transition to a license renewal period, the Low Temperature Overpressure Protection (LTOP) System enable temperature (T_{enable}) must account for embrittlement-increased RT_{NDT} values and additional margins, such as instrument uncertainty, imposed by Code or regulatory requirements. These factors may cause unnecessary restrictions in the RCS pressure/temperature operating window, and may also adversely impact the design bases for plants that require diverse means of low temperature overpressure relief using the residual heat removal (RHR) system relief valves. The RHR system is typically not designed for service above RCS temperatures of 350°F. Also, for plants which are required to operate shutdown cooling or decay heat removal systems at and below 300°F, T_{enable} values in this range increase complexity for the operators. As a means of maintaining acceptable margins of safety, satisfying the system licensing and design basis, and minimizing operational complexity, this evaluation demonstrates a method and provides the technical basis for determination of plant specific T_{enable} values for PWRs.

Nomenclature

F = Safety margin on pressure for determination of T_{enable} temperature
K_{Ia} = Critical arrest stress intensity factor (ksi-in$^{1/2}$)
K_{Ic} = Critical initiation stress intensity factor (ksi-in$^{1/2}$)
$K_{Im} = M_m \bullet (pR_i/t)$
K_{IR} = ASME reference stress intensity factor (ksi-in$^{1/2}$)
K_{It} = thermal stress intensity
LTOP = Low Pressure Overpressure Protection
M_m = Membrane stress correction factor
p = reactor vessel internal pressure (ksi)
R_I = vessel inner radius (in.)
RT_{NDT} = material adjusted reference temperature
t = vessel wall thickness (in.)
T_{enable} = Temperature at which LTOP systems must be effective or enabled.

Introduction

NRC Branch Technical Position RSB 5-2 [D-1] was revised in 1988 to include guidance on determination of the enabling temperature for LTOP systems (USNRC, 1988). In the years since, T_{enable} (or $T_{effective}$, as it has been designated in a recent ASME Section XI Code action) has become widely believed in the nuclear industry to be a fundamental material property, defined strictly by a margin from the material adjusted reference temperature (RT_{NDT}). Contrary to this, T_{enable} is a derived parameter based on several factors, including material fracture toughness, RPV dimensions, and the membrane stress intensity acting upon a postulated RPV surface flaw. BTP RSB 5-2 [D-1] specifies T_{enable} as the water temperature corresponding to a metal temperature of RT_{NDT} + 90°F. Although a basis has not been formally published, the 90°F value added to RT_{NDT} for the purpose of T_{enable} temperature determination is believed to have been calculated using the arrest fracture toughness K_{Ia}, and a factor of safety of 1.0 on design pressure.

The factor of safety on design pressure used in the determination of a T_{enable} temperature must not be confused with the 100% or 110% of allowable pressure at a given temperature permitted by ASME Section XI [D-2] for an LTOP pressure setpoint, depending on the reference fracture toughness used.

Gamble published the basis [D-3] for the definition of T_{enable} as RT_{NDT} + 50°F following the development of ASME Section XI Code Case N-514 [D-4]. This derivation of T_{enable} is based on determination of the temperature that would allow RCS pressure in a large 4-loop RPV to reach 110% of the reactor vessel design pressure without initiation of the ASME Section XI maximum postulated flaw. Again, this factor of safety on design pressure for T_{enable} temperature determination is not related to the 100% or 110% of allowable pressure permitted for an LTOP pressure setpoint at a given temperature, which depends on the reference fracture toughness used.

The N-514 basis document [D-3] further demonstrates that T_{enable} is dependent upon the following parameters:

a) Irradiation embrittlement adjusted reference temperature (RT_{NDT});
b) Vessel dimensions (inside radius and thickness exclusive of cladding);
c) Reference stress intensity factor (K_{Ic} or K_{Ia});
d) Pressure stress intensity; and
e) Safety margin provided on pressure stress intensity (1.0, 1.1, or 2.0).

This technical basis can be applied to calculate T_{enable} on a plant specific basis.

Making Margins Of Safety Consistent

Another benefit that can be achieved by determination of plant specific T_{enable} values is the application of a consistent margin of safety to all PWRs for this parameter. The definitions for enable temperature currently in use, as specified in N-514 [D-4] (which was incorporated into the 1993 Addenda of ASME Section XI Appendix G) or BTP RSB 5-2 [D-1], result in inconsistent margins of safety for PWRs. This is because T_{enable} is dependent upon reactor vessel dimensions, and reactor vessels that are smaller than the reference case for N-514 (e.g., all Westinghouse designed 2-loop and 3-loop reactors) are penalized when using T_{enable} criteria established for protection of larger reactor pressure vessels.

With the publication of ASME Section XI Code Case N-588 [D-5], another parameter affecting T_{enable} was identified. Code Case N-588 allowed the reference flaw applied to circumferential welds to be oriented circumferentially rather than axially. The Code Case takes credit for the extremely low likelihood of a flaw being oriented in an axial manner within circumferential weldments. This results in another inconsistency in T_{enable} margin of safety when this Code Case is applied, due to the effect of flaw orientation on allowable pressure. Because the currently defined values for T_{enable} are based on the stress intensity on an axially oriented reference flaw, in plants where N-588 [D-5] is applied, the current definitions for T_{enable} are inadequate.

The solution to the issue of inconsistent margin of safety is to develop and implement a method for determination of T_{enable} on a plant specific basis for any given pressurized water reactor vessel. This methodology will consider the factors identified above, most notably reactor vessel dimensions and postulated flaw orientation, and can be used to derive T_{enable} for each PWR vessel with a consistent and well defined margin of safety against brittle failure at low temperatures.

Design Basis For LTOPS Enable Temperature

The design bases for T_{enable} as defined in the basis document for N-514 [D-4] will be examined to document the assumptions and margins of safety implicit in this parameter. With this understanding, a plant specific approach to T_{enable} will be defined using a consistent design basis, such that equivalent and consistent margins of safety are established for all PWR reactor vessels.

The basis document [D-3] for Code Case N-514 [D-4] defines the basis for the LTOP enabling temperature as:

> "The LTOP enabling temperature assessment involved determining the temperature that would allow the pressure to reach 110% of the design pressure, or typically about 2,750 psi for PWRs, without initiation of a postulated quarter-thickness depth flaw having RT_{NDT} at the tip of the flaw equal to 300°F. . . . The results are presented in Figure 3 and indicate that pressure greater than 110% of design pressure is achieved at a temperature equal to approximately RT_{NDT} + 50°F."

It should be noted that the statement "initiation of a postulated flaw," implies that initiation fracture toughness, K_{Ic}, is utilized in this evaluation, in lieu of use of arrest fracture toughness, K_{Ia}. In fact, the Figure 3 that is referenced in the N-514 basis document notes that "Toughness = ASME K_{Ic}."

The N-514 basis document does not provide the specific underlying equations used to derive T_{enable}. However, using the information provided in the Code Case, it is possible to derive an explicit closed form solution for T_{enable}. This is provided below:

Derivation of Enabling Temperature – Code Case N-514

Based on ASME Section XI, G-2215 [D-2]:

$$K_{IR} > F \bullet K_{Im} + K_{It} \tag{1}$$

where:

- K_{IR} = ASME reference stress intensity factor (ksi-in$^{1/2}$)
- F = Safety margin on pressure for T_{enable} temperature determination
- $K_{Im} = M_m \bullet (p\, R_I / t)$
- K_{It} = 0, assuming isothermal conditions for LTOP
- M_m = Membrane stress correction factor from ASME Section III, Figure G-2214-1 (prior to 1996 Addenda)
- p = internal pressure (ksi)
- R_I = vessel inner radius (in.)
- t = vessel wall thickness (in.)

In the basis document for N-514, the following parameters were selected:

- $K_{Ic} = 33.2 + 20.734 \exp[0.02 (T - RT_{NDT})]$ is substituted for K_{IR} (the equation shown for K_{Ic} is taken from ASME Section XI Appendix A, Article A-4200)
- F = 1.1
- p = 2.5 ksia
- R_I = 86.9 inch
- t = 8.9 inch
- M_m = 2.87, ASME Section III Figure G-2214-1 (t = 8.9 inch, σ/σ_y = 0.5)

Substituting:

$$33.2 + 20.734 \exp[0.02 (T - RT_{NDT})] = [1.1 \bullet 2.87 \bullet 2.5 \bullet 86.9] / 8.9 \tag{2}$$

Which can be solved for T:

$$T = RT_{NDT} + 37.5°F \tag{3}$$

Based on engineering judgement, additional margin was added to this result to establish:

$$T_{enable} = RT_{NDT} + 50°F \tag{4}$$

Because the derivation of T_{enable} provided in the N-514 basis document was performed by somewhat graphical means, including an additional margin by rounding to RT_{NDT} + 50°F was reasonable to ensure that adequate safety margin was provided. However, when T_{enable} is explicitly calculated using a closed form solution, this additional margin is not necessary; sufficient margin is derived from including the factor of 1.1 on pressure in the T_{enable} calculation. The margin on temperature provided by calculating the T_{enable} temperature as the temperature at which the allowable pressure is 110% of design pressure, can be illustrated by calculating the T_{enable} which would result at 100% of design pressure:

$$33.2 + 20.734 \exp [0.02 (T - RT_{NDT})] = [1.0 \bullet 2.87 \bullet 2.5 \bullet 86.9] / 8.9 \qquad (5)$$

Which can be solved for T:

$$T = RT_{NDT} + 28.8°F \qquad (6)$$

This results in a difference of 37.5 - 28.8, or 8.7°F. No additional margin on temperature is needed; the margin on pressure demonstrated in the N-514 basis document [D-3] when the maximum pressure allowed by the LTOP system is 110% of the allowable pressure based on ASME Section XI Appendix G is already substantial, between 1.7 and 2.0. Since LTOP events are essentially isothermal, this margin on temperature is simply good engineering practice.

This case may also be evaluated using Westinghouse 2-loop reactor vessel dimensions (R_i = 66.16 inches, t = 6.5 inches) at a temperature of RT_{NDT} + 37.5°F, then solved for F (the safety margin on pressure). This results in a safety margin on pressure of 126% (utilizing the older Code stress intensity factors). This is significant in that it demonstrates the inconsistency of margin of safety based on a single generic enable temperature: at the same enable temperature, a large 4-loop RPV is protected against initiation of brittle failure to 2750 psig (110%), while a 2-loop Westinghouse RPV is protected to 3150 psig (126%). This represents a significant operating margin penalty on 2-loop reactors.

Derivation Of Relation For Plant Specific Enable Temperature

Using the methodology of N-514 [D-4], it is possible to establish T_{enable} for any size RPV with a calculation using the methodology defined in the Code Case basis document [D-3]. In addition, axial and circumferential flaw orientation will be considered in this evaluation by application of N-588 [D-5].

Stress Intensity for a Postulated Surface Flaw

Based on ASME Section XI, G-2215 (1996):

$$K_{IR} > F \bullet K_{Im} + K_{It} \qquad (7)$$

where:

K_{IR} = ASME reference stress intensity factor (ksi-in$^{1/2}$)
F = Safety margin on pressure for determination of T_{enable} temperature
$K_{Im} = M_m (p R_I / t)$
K_{It} = 0, assuming isothermal conditions for LTOP
p = internal pressure (ksi)
R_I = vessel inner radius (in.)
t = vessel wall thickness (in.)

The following parameters are selected to establish T_{enable}:

> $K_{Ic} = 33.2 + 20.734 \exp [0.02 (T - RT_{NDT})]$ is substituted for K_{IR}
> (the equation shown for K_{Ic} is taken from ASME Section XI Appendix A, Article A-4200)
> F = 1.1 (basis for Code Case N-514 T_{enable} temperature)
> p = vessel design pressure

Substituting and reducing:

$$33.2 + 20.734 \exp [0.02 (T - RT_{NDT})] = [1.1 \bullet M_m (p R_I / t)] \quad (8)$$

$$20.734 \exp [0.02 (T - RT_{NDT})] = [1.1 \bullet M_m (p R_I / t)] - 33.2 \quad (9)$$

$$\exp [0.02 (T - RT_{NDT})] = [(1.1 \bullet M_m (p R_I / t)) - 33.2] / 20.734 \quad (10)$$

$$0.02 (T - RT_{NDT}) = \ln [((1.1 \bullet M_m (p R_I / t)) - 33.2) / 20.734] \quad (11)$$

$$T - RT_{NDT} = 50 \ln [((1.1 \bullet M_m (p R_I / t)) - 33.2) / 20.734] \quad (12)$$

$$T = RT_{NDT} + 50 \ln [((1.1 \bullet M_m (p R_I / t)) - 33.2)/20.734] \quad (13)$$

Equation 13 establishes a relationship for determination of T_{enable} on a plant specific basis for any size RPV, and accounts for alternate postulated flaw orientations through the factor, M_m.

Example Calculation Of Enable Temperature For a Typical Westinghouse 2-Loop Reactor

Applying the plant-specific methodology above (along with the most recently available stress intensity factors from N-588 for axial and circumferential flaws) to a typical Westinghouse 2-Loop reactor, the LTOP system would be effective at coolant temperatures less than the greatest value of T_{enable} determined for: 1) the most limiting axial flaw; 2) the most limiting circumferential flaw; and 3) 200°F.

Inside Surface Axial Flaw

Solve Equation 13 for Westinghouse 2-Loop reactor dimensions assuming an inside surface axial flaw:

> $M_m = 0.926\, t^{1/2}$ for IS axial flaw, $2 \le t^{1/2} \le 3.464$ (Code Case N-588)
> p = vessel design pressure = 2.5 ksia
> R_I = 66.16 inch
> t = 6.5 inch

$$T = RT_{NDT} + 50 \ln [(1.1 \bullet 0.926 \bullet 6.5^{1/2} \bullet (2.5 \bullet 66.16/6.5) - 33.2)/20.734] \quad (14)$$

$$T = RT_{NDT} + 23.1°F \qquad (15)$$

This result establishes the enable temperature based on a postulated axial flaw for a typical Westinghouse 2-Loop reactor vessel.

Inside Surface Circumferential Flaw

Solve Equation 13 for Westinghouse 2-Loop reactor dimensions assuming an inside surface circumferential flaw:

M_m = 0.443 $t^{1/2}$ for circumferential flaw, $2 \leq t^{1/2} \leq 3.464$ (Code Case N-588)
p = vessel design pressure = 2.5 ksia
R_I = 66.16 inch
t = 6.5 inch

$$T = RT_{NDT} + 50 \ln [(1.1 \cdot 0.443 \cdot 6.5^{1/2} \cdot (2.5 \cdot 66.16/6.5) - 33.2)/20.734] \qquad (16)$$

$$T = RT_{NDT} + 50 \ln [(31.6 - 33.2) / 20.734] \qquad (17)$$

Equation 17 cannot be solved for T because the logarithm of a negative number would need to be taken. On a physical basis, this result means that the asymptotic minimum initiation fracture toughness of reactor vessel steels at infinitely low temperatures, 33.2 ksi-in$^{1/2}$ (the constant term of the K_{Ic} equation), is greater than the stress intensity imposed on the circumferential reference flaw (31.6 ksi-in$^{1/2}$) calculated for a 2-loop vessel at 2,750 psia. This means that for a circumferentially oriented flaw, growth of the reference flaw cannot initiate.

Restating this conclusion, the minimum available initiation fracture toughness of reactor vessel steels at any temperature is always greater than the crack opening stress intensity on a circumferential reference flaw in a Westinghouse 2-loop reactor vessel at 110% of the design pressure, assuming isothermal conditions.

Based on this evaluation, Westinghouse 2-Loop reactor LTOP systems would be effective at coolant temperatures less than 200°F, or at coolant temperatures corresponding to a reactor vessel metal temperature less than RT_{NDT} + 23°F for the most limiting of plates, forgings, and axial welds, whichever is greater. In this example, circumferential welds would never be controlling.

Conclusion

This evaluation demonstrates the procedure for calculating T_{enable} on a plant specific basis using a methodology consistent with Appendix G of ASME Code Section XI. The procedure also provides consideration of alternate reference flaw orientation in accordance with Code Case N-588. This establishes T_{enable} such that an appropriate level of vessel protection against brittle failure is provided at low temperatures, while improving plant operating margins.

On this basis, allowing for a simplified bounding approach as well as an explicit plant-specific approach, ASME Section XI approved a Code Case to implement these procedures.

References

[D-1] U.S. Nuclear Regulatory Commission, 1988, NUREG 0800, US NRC Standard Review Plan Branch Technical Position RSB 5-2, "Overpressure Protection of Pressurized Water Reactors While Operating at Low Temperatures," Revision 1.

[D-2] ASME Boiler and Pressure Vessel Code Section XI, Appendix G, 1986 Edition.

[D-3] Gamble, R.M., 1994, "ASME Code Guidelines for PWR Low Temperature Overpressure Protection System Limits," PVP-Vol. 285, ASME, New York.

[D-4] ASME, 1993, Section XI Code Case N-514, "Low Temperature Overpressure Protection."

[D-5] ASME, 1997, Section XI Code Case N-588, "Alternative to Reference Flaw Orientation of Appendix G for Circumferential Welds in Reactor Vessels."

APPENDIX E

Justification for ASME Code Case N-640 Exemption Request

ASME Section XI Code Case N-640

For North Anna Units 1 and 2, ASME Code Case N-640 is utilized in the development of the proposed pressure/temperature limits. These revised P/T limits have been developed using the K_{Ic} fracture toughness curve shown on ASME XI, Appendix A, Figure A-4200-1, in lieu of the K_{Ia} fracture toughness curve of ASME Section XI, Appendix G, Figure G-2210-1, as the lower bound for fracture toughness. The effect of granting this exemption is to change the fracture toughness curve used for development of the P-T curves from K_{Ia} to K_{Ic}. The other margins involved with the ASME XI, Appendix G process of determining P-T limit curves remain unchanged. The unchanged margins are: 1) a flaw which is 1/4 vessel thickness in depth and 3/2 the vessel thickness in length, 2) safety factor of two on pressure stress for heatup and cooldown and a safety factor of 1.5 for testing, and 3) upper bound adjusted reference temperature (RT_{NDT}).

Use of the K_{Ic} curve in determining the lower bound fracture toughness in the development of P/T operating limit curves is more technically correct than the K_{Ia} curve. The K_{Ic} curve models the slow heatup and cooldown process of a reactor coolant system, with the fastest rate allowed being 100°F per hour. The rate of change of pressure and temperature is very low at low temperatures; therefore, the reactor vessel thermal stress is essentially nil for this transient. During development of Code Case N-640 and the accompanying Appendix G code change, the ASME Section XI, Working Group on Operating Plant Criteria (WGOPC), performed assessments of the margins inherent to K_{Ia} using realistic heatup and cooldown curves. These assessments led to the conclusion that utilization of the K_{Ia} curve was excessively conservative and the K_{Ic} curve provided adequate margin for protection from brittle fracture. Technical bases for Code Case N-640 have been developed by the ASME Section XI Working Group on Plant Operating Criteria. Code Case N-640 was approved by Section XI and the ASME Main Committee, and was published in May 1999, and included in the 1999 Addenda to the ASME Code.

Use of this approach is justified by the initial conservatism of the K_{Ia} curve when the curve was codified in 1974. The initial conservatism was necessary due to limited experience and knowledge of the fracture toughness of reactor pressure vessel materials over time and usage. The conservatism also provided margin thought to be necessary to cover uncertainties and a number of postulated but unquantified effects.

Since 1974, additional knowledge has been gained from examination and testing of reactor pressure vessels that had been subject to the effects of neutron embrittlement in both an operating and test environment. The K_{Ia} curve was based on 125 data points. The K_{Ic} curve is based on more than 1500 data points. The additional data has significantly reduced the uncertainties associated with embrittlement effects and reduced other uncertainties. The added data ensures the K_{Ic} curve adequately statistically bounds the data. The new information indicates the lower bound on fracture toughness provided by the K_{Ia} curve is extremely conservative and is well beyond the margin of safety required to protect the public health and safety from potential reactor pressure vessel failure.

P/T limit curves based on the K_{Ic} methodology will enhance overall plant safety by opening the P-T operating window with the greatest safety benefit in the region of low temperature operations. There are two primary safety benefits in opening the lower temperature operating window. The first safety benefit is a reduction in the likelihood of a challenge to RCS power operated relief valve during low temperature operations. The second safety benefit is increasing the allowable operating pressure such that an RCP may be started without impinging upon the RCP NPSH requirement. Adequate NPSH minimizes wear to the RCP impeller due to cavitation, thereby reducing maintenance and personnel radiation exposure.

Justification for ASME Code Case N-640 Exemption Request

The following information provides the basis for the exemption request to 10 CFR 50.60 for use of ASME Section XI Code Case N-640, "Alternative Fracture Toughness for Development of P-T Limit Curves for ASME Section XI, Division I," in lieu of 10 CFR 50, Appendix G.

10 CFR 50.12 Requirements: The requested exemption to allow use of ASME Code Case N-640 in conjunction with ASME XI, Appendix G to determine the pressure-temperature limits meets the criteria of 10 CFR 50.12 as discussed below. 10 CFR 50.12 states that the Commission may grant an exemption from requirements contained in 10 CFR 50 provided that:

1. The requested exemption is authorized by law: No law exists which precludes the activities covered by this exemption request. 10 CFR 50.60(b) allows the use of alternatives to 10 CFR 50, Appendices G and H when an exemption is granted by the Commission under 10 CFR 50.12.

2. The requested exemption does not present an undue risk to the public health and safety: The revised pressure/temperature (P/T) limits being proposed for North Anna Units 1 and 2 rely in part, on the requested exemption. These revised P/T limits have been developed using the K_{Ic} fracture toughness curve shown on ASME XI, Appendix A, Figure A-4200-1, in lieu of the K_{Ia} fracture toughness curve of ASME XI, Appendix G, Figure G-2210-1, as the lower bound for fracture toughness. The other margins involved with the ASME XI, Appendix G process of determining P/T limit curves remain unchanged.

   Use of the K_{Ic} curve in determining the lower bound fracture toughness in the development of P/T operating limits curve is more technically correct than the K_{Ia} curve. The K_{Ic} curve models the slow heat-up and cooldown process of a reactor vessel.

   Use of this approach is justified by the initial conservatism of the K_{Ia} curve when the curve was codified in 1974. This initial conservatism was necessary due to limited knowledge of reactor pressure vessel materials over time and usage. Since 1974, additional knowledge has been gained about the effects of usage on reactor pressure vessel materials. The additional knowledge demonstrates the lower bound on fracture toughness provided by the K_{Ia} curve is well beyond the margin of safety required to protect the public health and safety from potential reactor pressure vessel failure.

   P/T curves based on the K_{Ic} curves will enhance overall plant safety by opening the pressure/temperature operating window with the greatest safety benefit in the region of low temperature operations. The two primary safety benefits in opening the low temperature operating window is a reduction in the challenges to RCS power operated relief valves and minimization of RCP impeller wear cavitation wear.

3. The requested exemption will not endanger the common defense and security: The common defense and security are not endangered by this exemption request.

4. Special circumstances are present which necessitate the request for an exemption to the regulations of 10 CFR 50.60: Pursuant to 10 CFR 50.12(a)(2), the NRC will consider granting an exemption to the regulations if special circumstances are present. This exemption meets the special circumstances of paragraphs:

   (a)(2)(ii) - demonstrates the underlying purpose of the regulation will continue to be achieved; and
   (a)(2)(iii) - would result in undue hardship or other cost that are significant if the regulation is enforced.

   10 CFR 50.12(a)(2)(ii): ASME XI, Appendix G, provides procedures for determining allowable loading on the reactor pressure vessel and is specified for that purpose by 10 CFR 50, Appendix G. Application of these procedures in the determination of P/T operating and test curves satisfied the underlying requirement for: 1) The reactor coolant pressure boundary be operated in a regime having sufficient margin to ensure, when stressed, the vessel boundary behaves in a non-brittle manner and the probability of a rapidly propagating fracture is minimized; and 2) P/T operating and test limit curves provide adequate margin in consideration of uncertainties in determining the effects of irradiation on material properties.

   The ASME XI, Appendix G, procedure was conservatively developed based on the level of knowledge existing in 1974 concerning reactor pressure vessel materials and the estimated effects of operation. Since 1974, the level of knowledge about these topics has been greatly expanded. This increased knowledge permits relaxation of the ASME XI, Appendix G, requirements via application of ASME Code Case N-640, while maintaining the underlying purpose of the ASME Code and the NRC regulations to ensure an acceptable margin of safety.

   10 CFR 50.12(a)(2)(iii): The Reactor Coolant System pressure/temperature operating window is defined by the P/T operating and test limit curves developed in accordance with the ASME XI, Appendix G procedure. Continued operation of North Anna Units 1 and 2 with these P/T curves without the relief provided by ASME Code Case N-640 would unnecessarily restrict the pressure-temperature operating window. This restriction diminishes pressure margin to the RCP NPSH requirement, potentially resulting in undesirable degradation of reactor coolant pump impellers due to cavitation.

   This constitutes an unnecessary burden that can be alleviated by the application of Code Case N-640 in the development of the proposed P/T curves. Implementation of the proposed P/T curves as allowed by ASME Code Case N-640 does not significantly reduce the margin of safety.

Code Case N-640, Conclusion for Exemption Acceptability: Compliance with the specified requirements of 10 CFR 50.60 would result in hardship and unusual difficulty without a compensating increase in the level of quality and safety. ASME Code Case N-640 allows a reduction in the fracture toughness lower bound used by ASME XI, Appendix G, in the determination of reactor coolant pressure-temperature limits. This proposed alternative is acceptable because the Code Case maintains the relative margin of safety commensurate with that which existed at the time ASME XI, Appendix G, was approved in 1974. Therefore, application of Code Case N-640 for North Anna Units 1 and 2 will ensure an acceptable margin of safety. The approach is justified by consideration of the overpressurization design basis events and the resulting margin to reactor vessel failure.

Restrictions on allowable operating conditions and equipment operability requirements have been established to ensure operating conditions are consistent with the assumptions of the accident analysis. Specifically, RCS pressure and temperature must be maintained within the heatup and cooldown rate dependent pressure-temperature limits specified in North Anna Units 1 and 2 Technical Specification 3.4.9.1 . Therefore, this exemption does not present an undue risk to the public health and safety.

APPENDIX F

Justification for Plant-Specific T_{enable} Exemption Request

Justification for Plant-Specific T_{enable} Exemption Request

The following information provides the basis for the exemption request to 10 CFR 50.60 for plant-specific implementation of the analysis methodology that supports ASME Section XI Code Case N-514, "Low Temperature Overpressure Protection Section XI, Division 1," in lieu of 10 CFR 50, Appendix G.

10 CFR 50.12 Requirements: The requested exemption to allow plant-specific implementation of the analytical method that supports ASME Code Case N-514 to determine the LTOPS enable temperature meets the criteria of 10 CFR 50.12 as addressed below. 10 CFR 50.12 states that the Commission may grant an exemption from requirements contained in 10 CFR 50 provided that:

1. The requested exemption is authorized by law: No law exists which precludes the activities covered by this exemption request. 10 CFR 50.60(b) allows the use of alternatives to 10 CFR 50, Appendices G and H when an exemption is granted by the Commission under 10 CFR 50.12.

2. The requested exemption does not present an undue risk to the public health and safety: A revised analysis basis for the North Anna Units 1 and 2 Technical Specification LTOPS enable temperature (T_{enable}) is proposed. The analysis basis for T_{enable} has been developed to provide bounding reactor vessel low temperature integrity protection during the LTOPS design basis transients. The LTOPS PORV lift setpoint utilizes 100% of the pressure determined to satisfy Appendix G, paragraph G-2215 of ASME Section XI, Division 1, as a design limit. The approach is justified by consideration of the overpressurization design basis events and the resulting margin to reactor vessel failure. Restrictions on allowable operating conditions and equipment operability requirements have been established to ensure that operating conditions are consistent with the assumptions of the accident analysis. Therefore, this exemption does not present an undue risk to the public health and safety.

3. The requested exemption will not endanger the common defense and security: The common defense and security are not endangered by this exemption request.

4. Special circumstances are present which necessitate the request for an exemption to the regulations of 10 CFR 50.60: Pursuant to 10 CFR 50.12(a)(2), the NRC will consider granting an exemption to the regulations if special circumstances are present. This exemption meets the special circumstances of paragraphs:

   (a)(2)(ii) - demonstrates that the underlying purpose of the regulation will continue to be achieved, and
   (a)(2)(iii) - would result in undue hardship or other cost that are significant if the regulation is enforced.

10 CFR 50.12(a)(2)(ii): ASME Code Case N-514 recognizes the conservatism of the ASME XI, Appendix G curves and allows setting the LTOPS PORV lift setpoints such that 110% of the ASME Section XI, Appendix G limits are not exceeded. Since Code Case N-514 is used in conjunction with Code Case N-640, an LTOPS setpoint design limit of 100% of the ASME XI Appendix G limits is allowed. Code Case N-514 permits use of a generic LTOPS enable temperature equal to an [adjusted] RT_{NDT} + 50°F or 200 °F, whichever is greater for the limiting material. The generic formulation of RT_{NDT} + 50°F specified in Code Case N-514 is being replaced with a plant-specific formulation, since the plant-specific formulation maintains the margin of safety inherent in the generic formulation. The margin of safety inherent in Code Case N-514 is maintained by demonstrating that the fracture criterion employed in the development of the Code Case N-514 generic T_{enable} formulation is met on a plant-specific basis. Plant-specific application of the analysis methodology that supports Code Case N-514 permits implementation of LTOPS PORV lift setpoints that preserve an acceptable margin of safety while maintaining operational margins for reactor coolant pump operation at low temperatures and pressures. The LTOPS enable temperature established in accordance with a plant-specific implementation of the analysis methodology that supports ASME Code Case N-514 will also minimize the unnecessary actuation of protection system pressure relieving devices. Therefore, establishing the LTOP setpoint in accordance with ASME Code Case N-514 analysis criteria satisfies the underlying purpose of the ASME Code and the NRC regulations to ensure an acceptable level of safety.

10 CFR 50.12(a)(2)(iii): The Reactor Coolant System pressure-temperature operating window at low temperatures is defined by the LTOPS setpoints and T_{enable}. Implementation of an LTOPS T_{enable} without the additional margin associated with plant-specific implementation of the ASME Code Case N-514 analysis methodology would unnecessarily restrict the pressure-temperature operating window. Removal of the restriction would minimize RCP impeller cavitation wear while operating in the LTOPS region and would reduce the potential for undesired actuation of LTOPS PORVs. This constitutes an unnecessary burden that can be alleviated by plant-specific application of the analysis methodology that supports Code Case N-514. Implementation LTOPS T_{enable} values in accordance with the ASME Code Case N-514 analysis methodology does not significantly reduce the margin of safety associated with normal operational heatup and cooldown limits.

Code Case N-514, Conclusion for Exemption Acceptability: Compliance with the specified requirements of 10 CFR 50.60 would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety. Plant-specific implementation of the analysis methodology that supports ASME Code Case N-514 allows setting the LTOPS enable temperature such that bounding protection of ASME Section XI, Appendix G limits is provided. This proposed alternative is acceptable because the proposed methodology establishes LTOPS setpoints that retain an acceptable margin of safety, maintains adequate operational margins for reactor coolant pump operation at low temperatures and pressures, and minimizes the potential for an undesired LTOPS actuation. Therefore, plant-specific application of the analysis methodology that supports Code Case N-514 for North Anna Units 1 and 2 will ensure an acceptable level of safe.

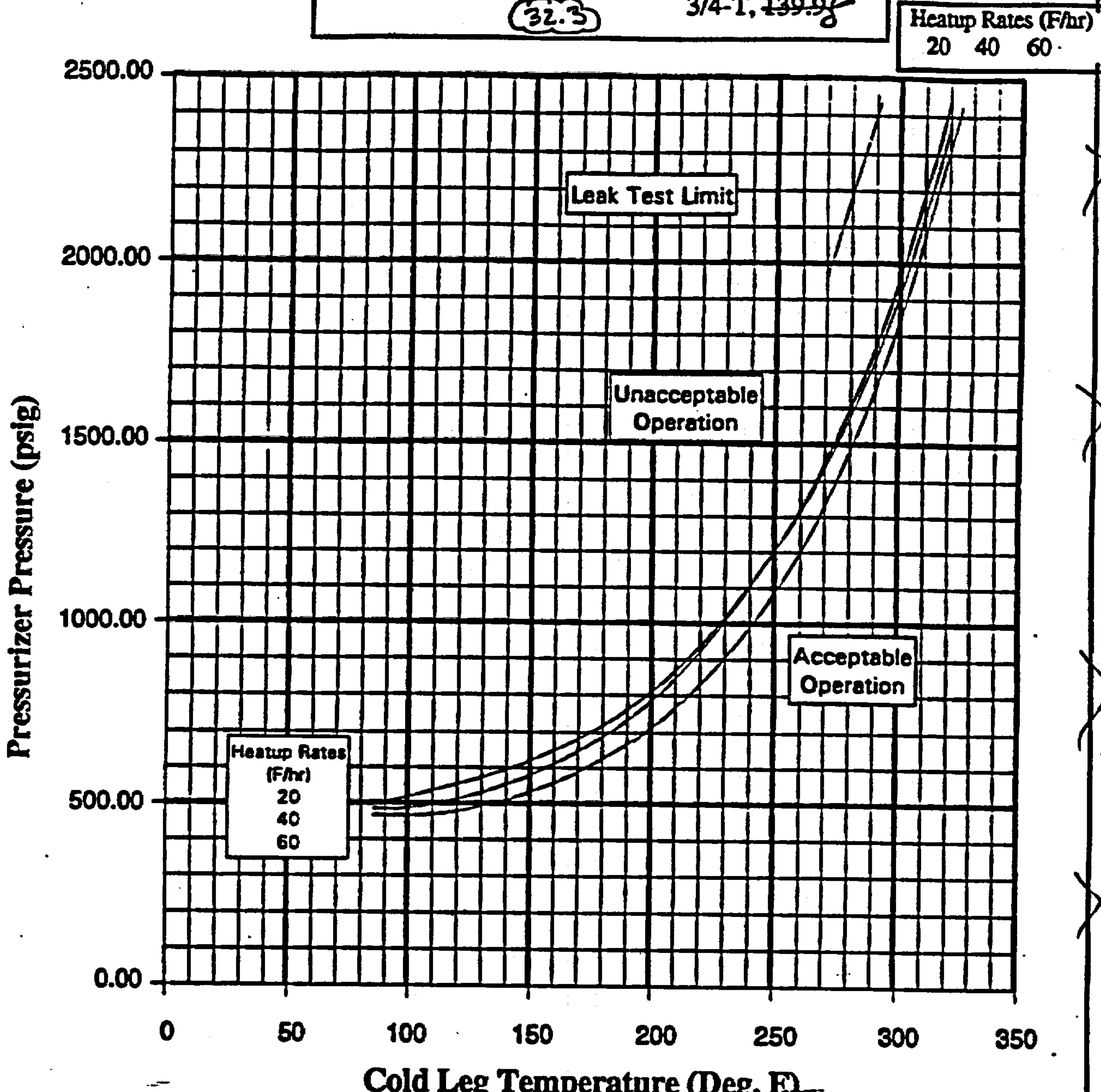
Attachment 2

Mark-up of Unit 1 and Unit 2 Technical Specifications Changes

North Anna Power Station
Units 1 and 2
Virginia Electric and Power Company

Figure 3.4-2 — North Anna Unit 1 Reactor Coolant System Heatup Limitations

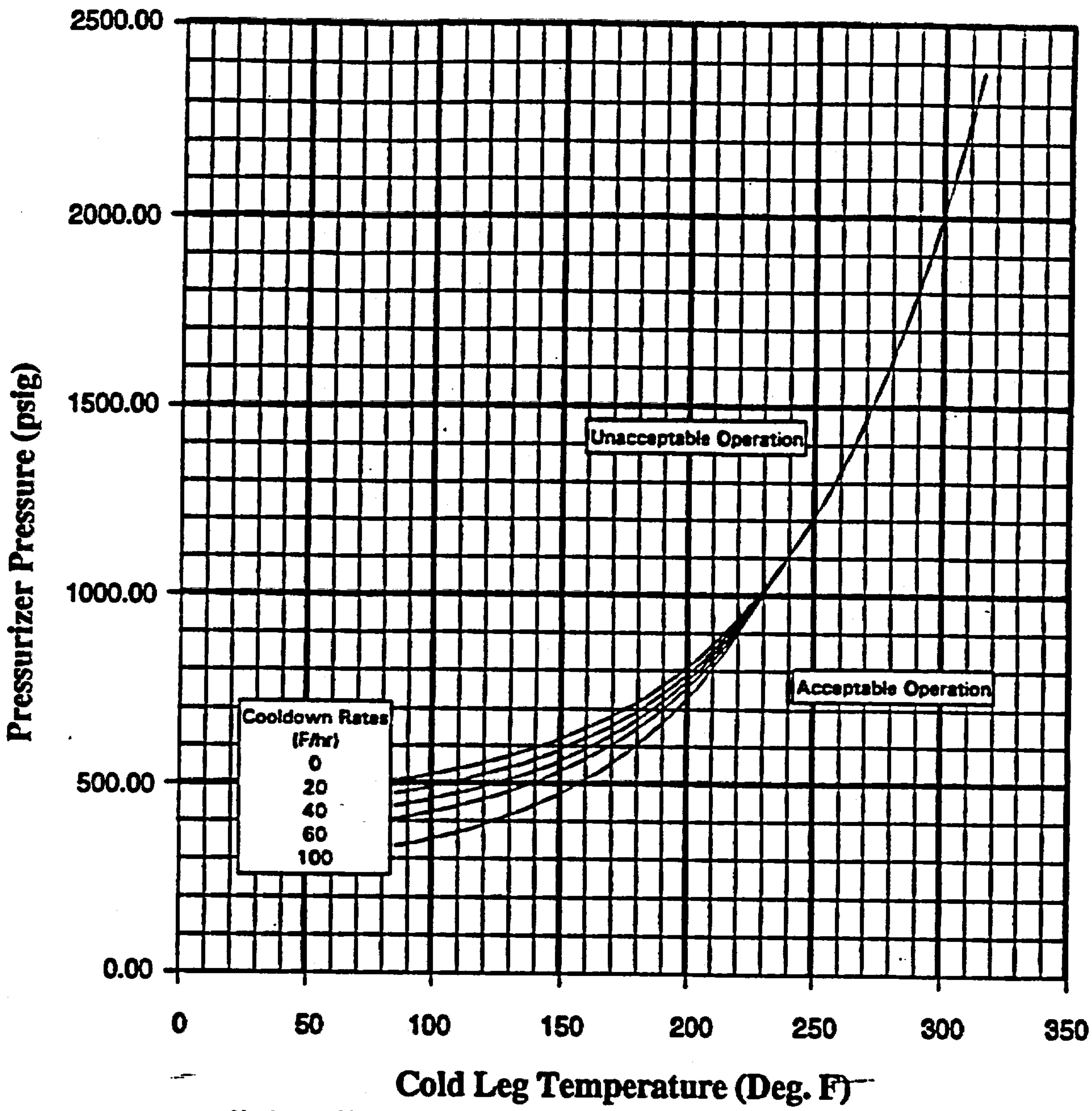
Material Property Basis
~~Limiting Material: Circumferential Weld Seam~~
Limiting ART at ~~30.7~~ 32.3 EFPY: 1/4-T, ~~162.9 F~~ 218.5°F
3/4-T, ~~139.9~~ 195.6°F



North Anna Unit 1 Reactor Coolant System Heatup Limitations (Heatup Rates up to 60 F/hr) Applicable for the First ~~30.7~~ 32.3 EFPY (Without Margins for Instrumentation Errors)

Figure 3.4-3 — North Anna Unit 1
Reactor Coolant System Cooldown Limitations

Material Property Basis
~~Limiting Material: Circumferential Weld Seam~~
Limiting ART at ~~30.7~~ 32.3 EFPY: 1/4-T, ~~162.9 F~~ 218.5°F
3/4-T, ~~139.9 F~~ 195.6°F



North Anna Unit 1 Reactor Coolant System Cooldown Limitations (Cooldown Rates up to 100 F/hr) Applicable for the First ~~30.7~~ 32.3 EFPY (Without Margins for Instrumentation Errors)

The heatup limit curve, Figure 3.4-2, is a composite curve which was prepared by determining the most conservative case, with either the inside or outside wall controlling, for any heatup rate up to 60°F per hour. The cooldown limit curves of Figure 3.4-3 are composite curves which were prepared based upon the same type analysis with the exception that the controlling location is always the inside wall where the cooldown thermal gradients tend to produce tensile stresses while producing compressive stresses at the outside wall. ~~The heatup and cooldown curves were prepared based upon the most limiting value of the predicted adjusted reference temperature at the end of 30.7 EFPY. The most recent capsule analysis results are documented in Westinghouse Report WCAP-11777, February 1988. The heatup and cooldown curves are documented in Westinghouse Report WCAP-13831, Rev. 1, August 1993.~~

Insert 1.

The reactor vessel materials have been tested to determine their initial RT_{NDT}. Reactor operation and resultant fast neutron (E > 1 Mev) irradiation will cause an increase in the RT_{NDT}. An adjusted reference temperature, based upon the fluence and copper content of the material in question, can be predicted using US NRC Regulatory Guide ~~1.98~~ 1.99, Revision 2. The heatup and cooldown limit curves (Figure 3.4-2 and Figure 3.4-3) include predicted adjustments for this shift in RT_{NDT} at the end of ~~30.7~~ 32.3 EFPY. ~~The reactor vessel beltline region material properties are listed on Figures 3.4-2 and 3.4-3.~~

The actual shift in the RT_{NDT} of the vessel material ~~will be~~ is established periodically by removal and evaluation of the reactor vessel material specimens installed on the inside wall of the thermal shield. The surveillance capsule withdrawal schedule was prepared in accordance with the requirements of ASTM E-185 and is presented in the UFSAR. Regulatory Guide 1.99, Revision 2, provides guidance for calculation of the shift in RT_{NDT} using measured data. Dosimetry from the surveillance capsule is used to ~~determine~~ provide benchmarks for calculation of the neutron fluence to which the material specimens ~~were exposed, and to support calculational estimates of the neutron fluence to the reactor vessel.~~ and the reactor vessel were exposed.

The pressure-temperature limit lines shown on Figure 3.4-2 for inservice leak and hydrostatic testing have been provided to assure compliance with the minimum temperature requirements of Appendix G to 10 CFR 50. The minimum temperature for criticality specified in T.S. 3.1.1.5 assures compliance with the criticality limits of 10 CFR 50 Appendix G.

The number of reactor vessel irradiation surveillance specimens and the frequencies for removing and testing these specimens are provided in the UFSAR to assure compliance with the requirements of Appendix H to 10 CFR Part 50.

Pressurizer

The limitations imposed on pressurizer heatup and cooldown and spray water temperature differential are provided to assure that the pressurizer is operated within the design criteria assumed for the fatigue analysis performed in accordance with the ASME Code requirements.

INSERT 1

The heatup and cooldown curves of Figures 3.4-2 and 3.4-3 conservatively bound the design basis heatup and cooldown curves, which were based upon a 1/4-T RT_{NDT} value of 218.5°F and a 3/4-T RT_{NDT} value of 195.6°F. These RT_{NDT} values conservatively bound the predicted reactor vessel beltline RT_{NDT} values for North Anna Unit 1 operation through 32.3 EFPY.

Low-Temperature Overpressure Protection

The OPERABILITY of two PORVs or an RCS vent opening of greater than 2.07 square inches ensures that the RCS will be protected from pressure transients which could exceed the limits of Appendix G to 10 CFR Part 50 when one or more of the RCS cold legs are less than or equal to 235°F. Either PORV has adequate relieving capability to protect the RCS from overpressurization when the transient is limited to either (1) the start of an idle RCP with the secondary water temperature of the steam generator less than or equal to 50°F above the RCS cold leg temperatures or (2) the start of a charging pump and its injection into a water-solid RCS. INSERT 2

Automatic or passive low temperature overpressure protection (LTOP) is required whenever any RCS cold leg temperature is less than 235°F. This temperature ~~is~~ conservatively bounds the water temperature corresponding to a metal temperature of ~~at least~~ the limiting RT_{NDT} + ~~50°F~~ 31.9°F + instrument uncertainty. Above 235°F administrative control is adequate protection to ensure the limits of the heatup curve (Figure 3.4-2) and the cooldown curve (Figure 3.4-3) are not violated. The concept of requiring automatic LTOP at the lower end, and administrative control at the upper end, of the Appendix G curves is further discussed in NRC Generic Letter 88-11.

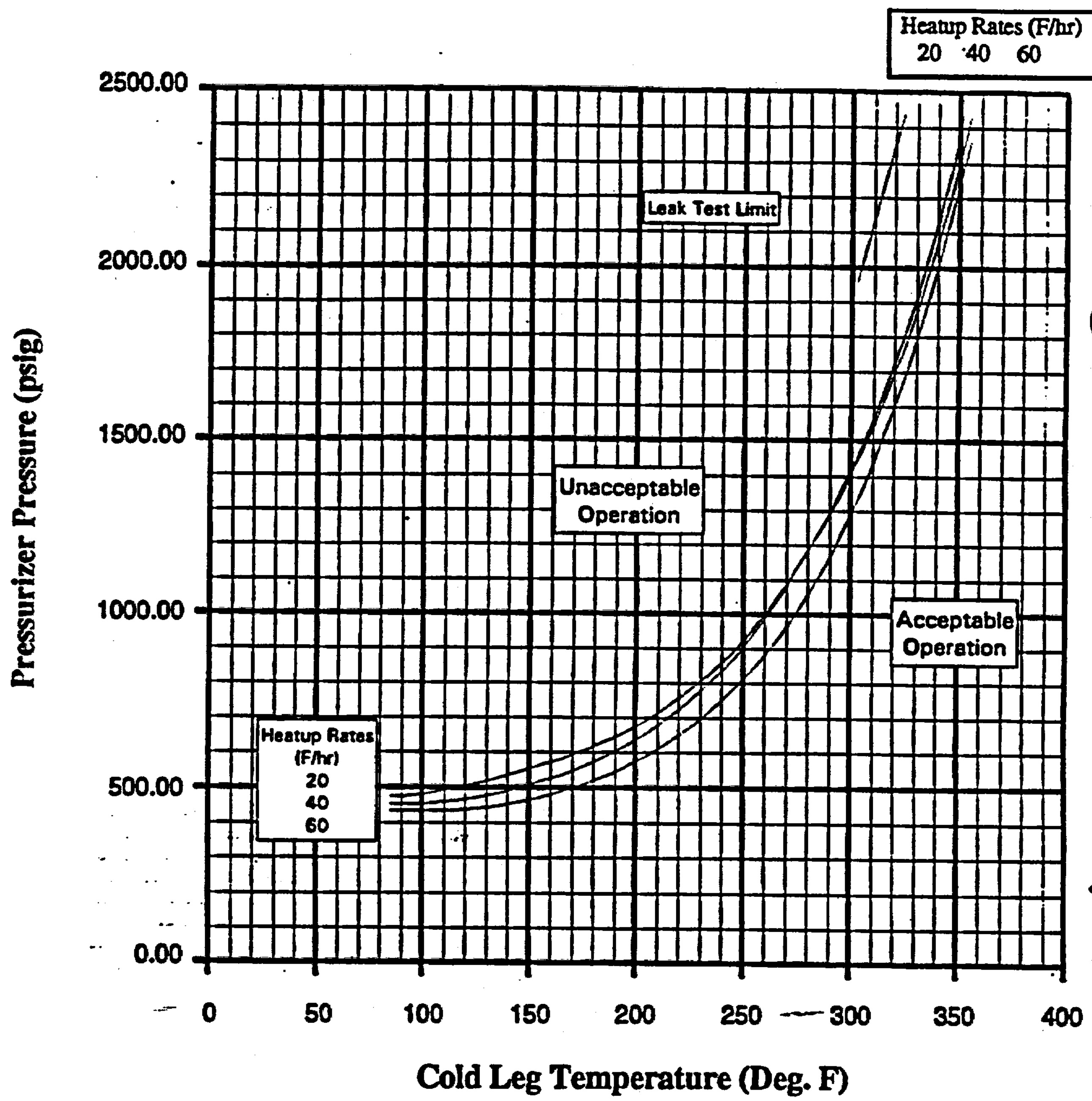
Surveillance limits are established for the pressure in the backup nitrogen accumulators to ensure there is adequate motive power for the PORVs to cope with an inadvertent start of a high head safety injection pump in a water solid condition, allowing adequate time for the operators to respond to terminate the event.

INSERT 2

The low temperature PORV lift setpoints were established to ensure that pressure at the reactor vessel beltline during these design basis events will not exceed 100% of the 10 CFR 50 Appendix G isothermal limit curve when the LTOP system is enabled.

Figure 3.4-2 — North Anna Unit 2 Reactor Coolant System Heatup Limitations

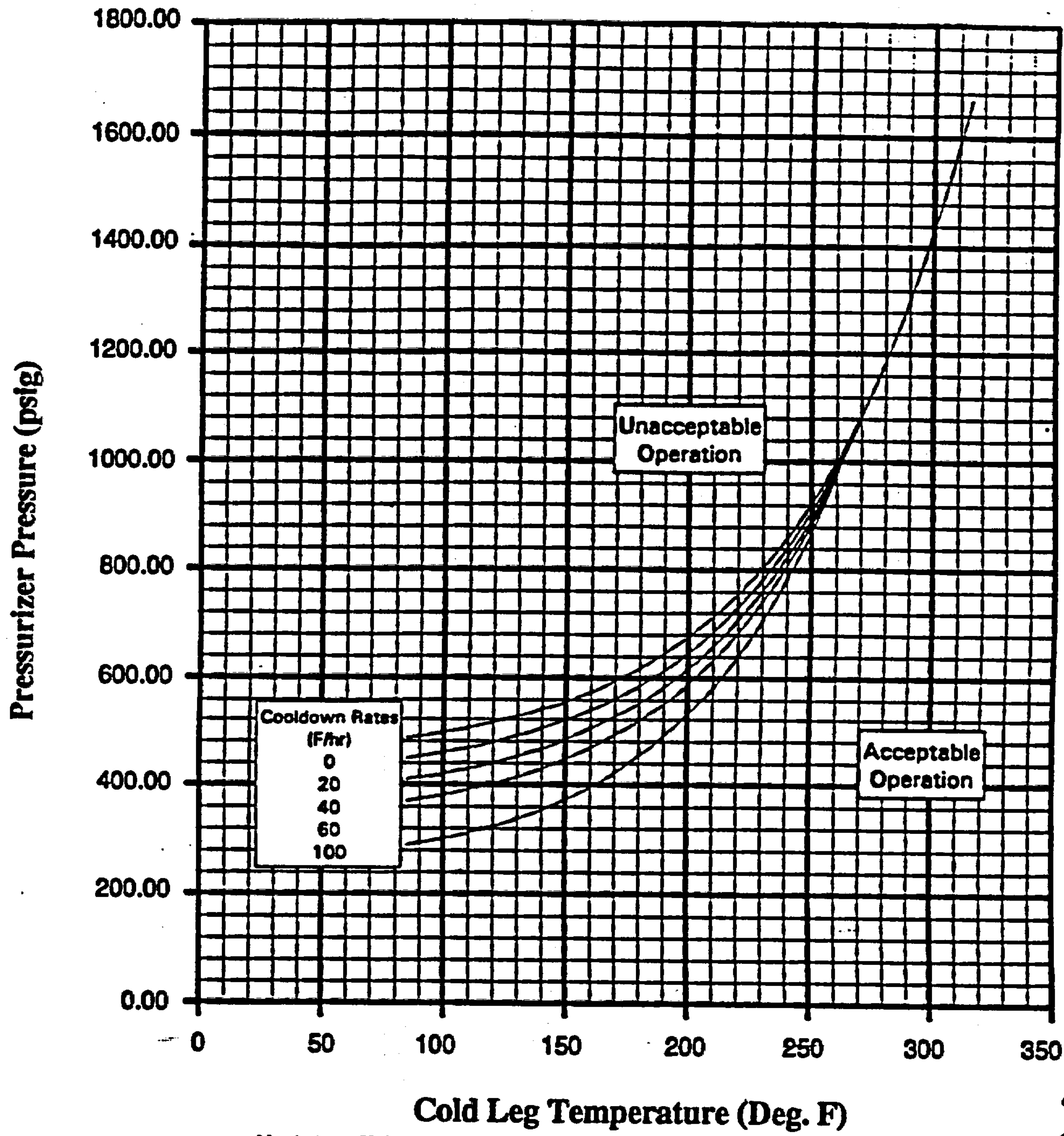
Material Property Basis	
~~Limiting Material: Lower Shell Plate~~	
Limiting ART at ~~17~~ 34.3 EFPY:	1/4-T, ~~196 F~~ 218.5°F
	3/4-T, ~~172 F~~ 195.6°F



North Anna Unit 2 Reactor Coolant System Heatup Limitations (Heatup Rates up to 60 F/hr) Applicable for the First ~~17~~ 34.3 EFPY (Without Margins for Instrumentation Errors)

Figure 3.4-3 — North Anna Unit 2
Reactor Coolant System Cooldown Limitations

Material Property Basis	
~~Limiting Material: Lower Shell Plate~~	
Limiting ART at ~~17~~ 34.3 EFPY:	1/4-T, ~~196 F~~ 218.5°F
	3/4-T, ~~172 F~~ 195.6°F



North Anna Unit 2 Reactor Coolant System Cooldown Limitations (Cooldown Rates up to 100 F/hr) Applicable for the First ~~17~~ 34.3 EFPY (Without Margins for Instrumentation Errors)

INSERT 3

The heatup limit curve, Figure 3.4-2, is a composite curve which was prepared by determining the most conservative case, with either the inside or outside wall controlling, for any heatup rate up to 60°F per hour. The cooldown limit curves of Figure 3.4-3 are composite curves which were prepared based upon the same type analysis with the exception that the controlling location is always the inside wall where the cooldown thermal gradients tend to produce tensile stresses while producing compressive stresses at the outside wall. ~~The heatup and cooldown curves were prepared based upon the most limiting value of the predicted adjusted reference temperature at the end of 17 EFPY. The most recent capsule analysis results are documented in Westinghouse Reports WCAP-12497, January 1990. The heatup and cooldown curves are documented in Westinghouse Report WCAP-12503, March, 1990.~~

1.99

The reactor vessel materials have been tested to determine their initial RT_{NDT}. Reactor operation and resultant fast neutron (E > 1 Mev) irradiation will cause an increase in the RT_{NDT}. An adjusted reference temperature, based upon the fluence and copper content of the material in question, can be predicted using US NRC Regulatory Guide ~~1.98~~, Revision 2. The heatup and cooldown limit curves (Figure 3.4-2 and Figure 3.4-3) include predicted adjustments for this shift in RT_{NDT} at the end of ~~17~~ 34.3 EFPY. ~~The reactor vessel beltline region material properties are listed on Figures 3.4-2 and 3.4-3.~~

The actual shift in the RT_{NDT} of the vessel material ~~will be~~ is established periodically by removal and evaluation of the reactor vessel material specimens installed on the inside wall of the thermal shield. The surveillance capsule withdrawal schedule was prepared in accordance with the requirements of ASTM E-185 and is presented in the UFSAR. Regulatory Guide 1.99, Revision 2, provides guidance for calculation of the shift in RT_{NDT} using measured data. Dosimetry from the surveillance capsule is used to ~~determine~~ provide benchmarks for calculation of the neutron fluence to which the material specimens and the reactor vessel were exposed ~~were exposed, and to support calculational estimates of the neutron fluence to the reactor vessel.~~

The pressure-temperature limit lines shown on Figure 3.4-2 for inservice leak and hydrostatic testing have been provided to assure compliance with the minimum temperature requirements of Appendix G to 10 CFR 50. The minimum temperature for criticality specified in T.S. 3.1.1.5 assures compliance with the criticality limits of 10 CFR 50 Appendix G.

The number of reactor vessel irradiation surveillance specimens and the frequencies for removing and testing these specimens are provided in the UFSAR to assure compliance with the requirements of Appendix H to 10 CFR Part 50.

Pressurizer

The limitations imposed on pressurizer heatup and cooldown and spray water temperature differential are provided to assure that the pressurizer is operated within the design criteria assumed for the fatigue analysis performed in accordance with the ASME Code requirements.

INSERT 3

The heatup and cooldown curves of Figures 3.4-2 and 3.4-3 conservatively bound the design basis heatup and cooldown curves, which were based upon a 1/4-T RT_{NDT} value of 218.5°F and a 3/4-T RT_{NDT} value of 195.6°F. These RT_{NDT} values conservatively bound the predicted reactor vessel beltline RT_{NDT} values for North Anna Unit 2 operation through 34.3 EFPY.

Low-Temperature Overpressure Protection

The OPERABILITY of two PORVs or an RCS vent opening of greater than 2.07 square inches ensures that the RCS will be protected from pressure transients which could exceed the limits of Appendix G to 10 CFR Part 50 when one or more of the RCS cold legs are less than or equal to 270°F. Either PORV has adequate relieving capability to protect the RCS from overpressurization when the transient is limited to either (1) the start of an idle RCP with the secondary water temperature of the steam generator less than or equal to 50°F above the RCS cold leg temperatures or (2) the start of a charging pump and its injection into a water-solid RCS. INSERT 4

Automatic or passive low temperature overpressure protection (LTOP) is required whenever any RCS cold leg temperature is less than 270°F. This temperature ~~is~~ conservatively bounds the water temperature corresponding to a metal temperature of ~~at least~~ the limiting RT_{NDT} + ~~50°F~~ 31.9°F + instrument uncertainty. Above 270°F administrative control is adequate protection to ensure the limits of the heatup curve (Figure 3.4-2) and the cooldown curve (Figure 3.4-3) are not violated. The concept of requiring automatic LTOP at the lower end, and administrative control at the upper end, of the Appendix G curves is further discussed in NRC Generic Letter 88-11.

Surveillance limits are established for the pressure in the backup nitrogen accumulators to ensure there is adequate motive power for the PORVs to cope with an inadvertent start of a high head safety injection pump in a water solid condition, allowing adequate time for the operators to respond to terminate the event.

3/4.4.10 STRUCTURAL INTEGRITY

3/4.4.10.1 ASME CODE CLASS 1, 2 and 3 COMPONENTS

The inspection programs for ASME Code Class 1, 2 and 3 Reactor Coolant System components ensure that the structural integrity of these components will be maintained at an acceptable level throughout the life of the plant. To the extent applicable, the inspection program for components is in compliance with Section XI of the ASME Boiler and Pressure Vessel Code.

INSERT 4

The low temperature PORV lift setpoints were established to ensure that pressure at the reactor vessel beltline during these design basis events will not exceed 100% of the 10 CFR 50 Appendix G isothermal limit curve when the LTOP system is enabled.

Attachment 3

Proposed Unit 1 and Unit 2 Technical Specifications Changes

North Anna Power Station
Units 1 and 2
Virginia Electric and Power Company

UNIT 1

TECH SPEC CHANGE REQUEST NO. 376

TABULATION OF CHANGES

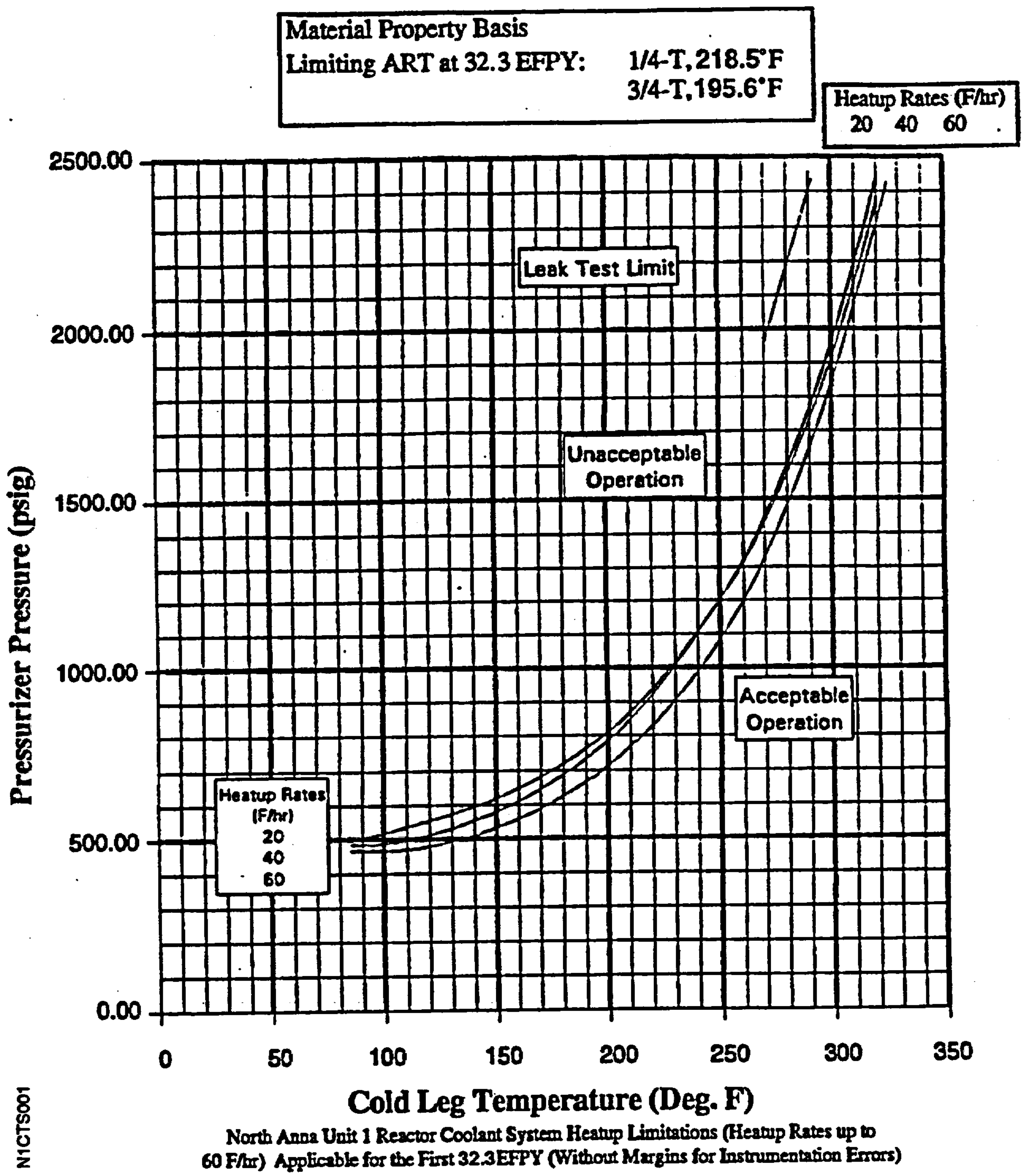
License No. NPF-4 / Docket No. 50-338

Summary of change:

This proposed change to the Technical Specifications is being made to extend the cumulative core burnup applicability limits for the P/T limits, LTOPS setpoints, and T_{enable} values.

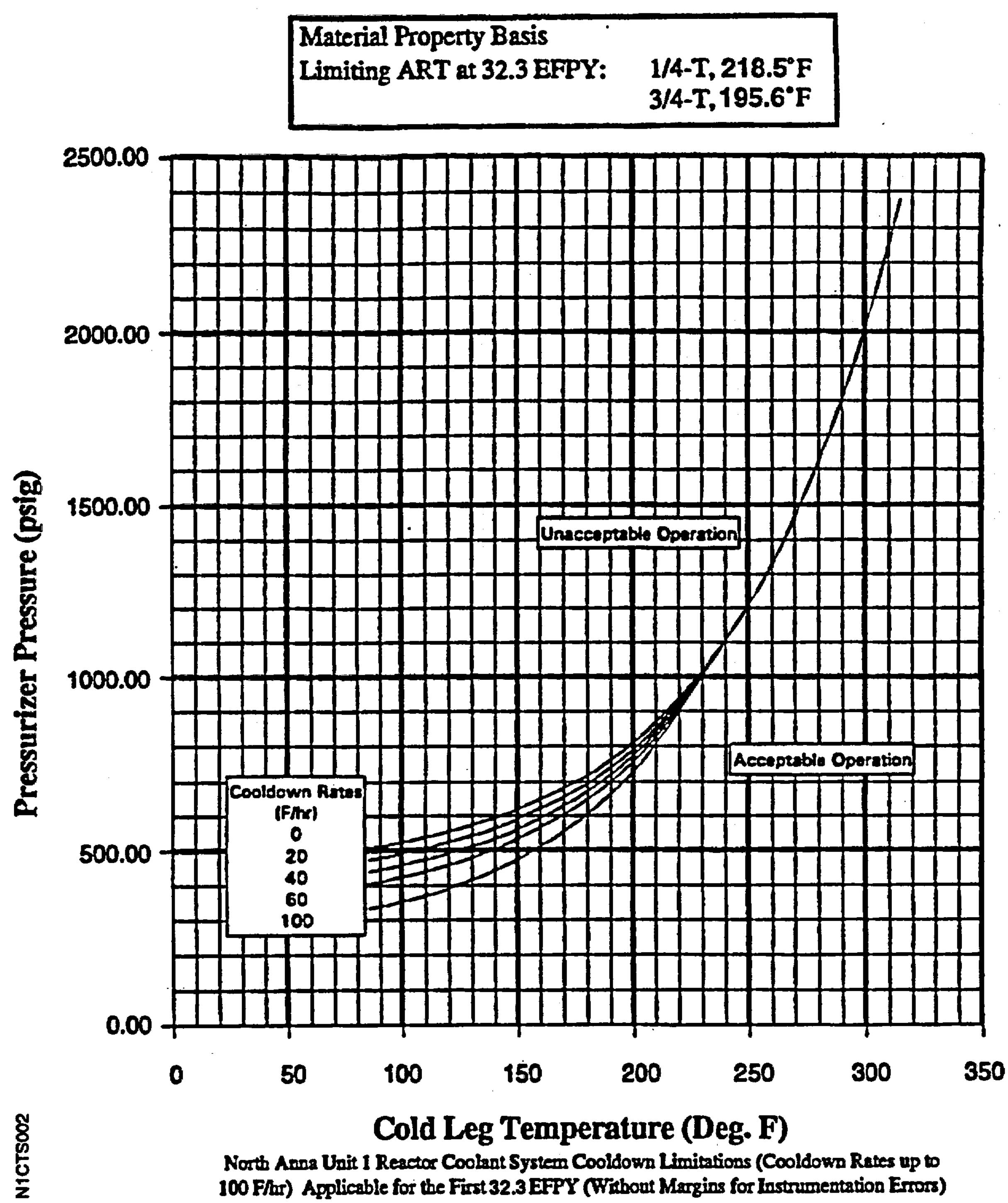
DELETE	DATED	SUBSTITUTE
3/4 4-27	10-05-94	3/4 4-27
3/4 4-28	10-05-94	3/4 4-28
B 3/4 4-7	10-05-94	B 3/4 4-7
B 3/4 4-8	03-02-99	B 3/4 4-8

Figure 3.4-2 — **North Anna Unit 1**
Reactor Coolant System Heatup Limitations



North Anna Unit 1 Reactor Coolant System Heatup Limitations (Heatup Rates up to 60 F/hr) Applicable for the First 32.3EFPY (Without Margins for Instrumentation Errors)

Figure 3.4-3 — North Anna Unit 1 Reactor Coolant System Cooldown Limitations



North Anna Unit 1 Reactor Coolant System Cooldown Limitations (Cooldown Rates up to 100 F/hr) Applicable for the First 32.3 EFPY (Without Margins for Instrumentation Errors)

The heatup limit curve, Figure 3.4-2, is a composite curve which was prepared by determining the most conservative case, with either the inside or outside wall controlling, for any heatup rate up to 60°F per hour. The cooldown limit curves of Figure 3.4-3 are composite curves which were prepared based upon the same type analysis with the exception that the controlling location is always the inside wall where the cooldown thermal gradients tend to produce tensile stresses while producing compressive stresses at the outside wall. The heatup and cooldown curves of Figures 3.4-2 and 3.4-3 conservatively bound the design basis heatup and cooldown curves, which were based upon a 1/4-T RT_{NDT} value of 218.5°F and a 3/4-T RT_{NDT} value of 195.6°F. These RT_{NDT} values conservatively bound the predicted reactor vessel beltline RT_{NDT} values for North Anna Unit 1 operation through 32.3 EFPY.

The reactor vessel materials have been tested to determine their initial RT_{NDT}. Reactor operation and resultant fast neutron (E > 1 Mev) irradiation will cause an increase in the RT_{NDT}. An adjusted reference temperature, based upon the fluence and copper content of the material in question, can be predicted using US NRC Regulatory Guide 1.99, Revision 2. The heatup and cooldown limit curves (Figure 3.4-2 and Figure 3.4-3) include adjustments for this predicted shift in RT_{NDT} at the end of 32.3 EFPY.

The actual shift in the RT_{NDT} of the vessel material is established periodically by removal and evaluation of the reactor vessel material specimens installed on the inside wall of the thermal shield. The surveillance capsule withdrawal schedule was prepared in accordance with the requirements of ASTM E-185 and is presented in the UFSAR. Regulatory Guide 1.99, Revision 2, provides guidance for calculation of the shift in RT_{NDT} using measured data. Dosimetry from the surveillance capsule is used to provide benchmarks for calculation of the neutron fluence to which the material specimens and the reactor vessel were exposed.

The pressure-temperature limit lines shown on Figure 3.4-2 for inservice leak and hydrostatic testing have been provided to assure compliance with the minimum temperature requirements of Appendix G to 10 CFR 50. The minimum temperature for criticality specified in T.S. 3.1.1.5 assures compliance with the criticality limits of 10 CFR 50 Appendix G.

The number of reactor vessel irradiation surveillance specimens and the frequencies for removing and testing these specimens are provided in the UFSAR to assure compliance with the requirements of Appendix H to 10 CFR Part 50.

Pressurizer

The limitations imposed on pressurizer heatup and cooldown and spray water temperature differential are provided to assure that the pressurizer is operated within the design criteria assumed for the fatigue analysis performed in accordance with the ASME Code requirements.

Low-Temperature Overpressure Protection

The OPERABILITY of two PORVs or an RCS vent opening of greater than 2.07 square inches ensures that the RCS will be protected from pressure transients which could exceed the limits of Appendix G to 10 CFR Part 50 when one or more of the RCS cold legs are less than or equal to 235°F. Either PORV has adequate relieving capability to protect the RCS from overpressurization when the transient is limited to either (1) the start of an idle RCP with the secondary water temperature of the steam generator less than or equal to 50°F above the RCS cold leg temperatures or (2) the start of a charging pump and its injection into a water-solid RCS. The low temperature PORV lift setpoints were established to ensure that pressure at the reactor vessel beltline during these design basis events will not exceed 100% of the 10 CFR 50 Appendix G isothermal limit curve when the LTOP system is enabled.

Automatic or passive low temperature overpressure protection (LTOP) is required whenever any RCS cold leg temperature is less than 235°F. This temperature conservatively bounds the water temperature corresponding to a metal temperature of the limiting RT_{NDT}+31.9°F + instrument uncertainty. Above 235°F administrative control is adequate protection to ensure the limits of the heatup curve (Figure 3.4-2) and the cooldown curve (Figure 3.4-3) are not violated. The concept of requiring automatic LTOP at the lower end, and administrative control at the upper end, of the Appendix G curves is further discussed in NRC Generic Letter 88-11.

Surveillance limits are established for the pressure in the backup nitrogen accumulators to ensure there is adequate motive power for the PORVs to cope with an inadvertent start of a high head safety injection pump in a water solid condition, allowing adequate time for the operators to respond to terminate the event.

UNIT 2

TECH SPEC CHANGE REQUEST NO. 376

TABULATION OF CHANGES

License No. NPF-7 / Docket No. 50-339

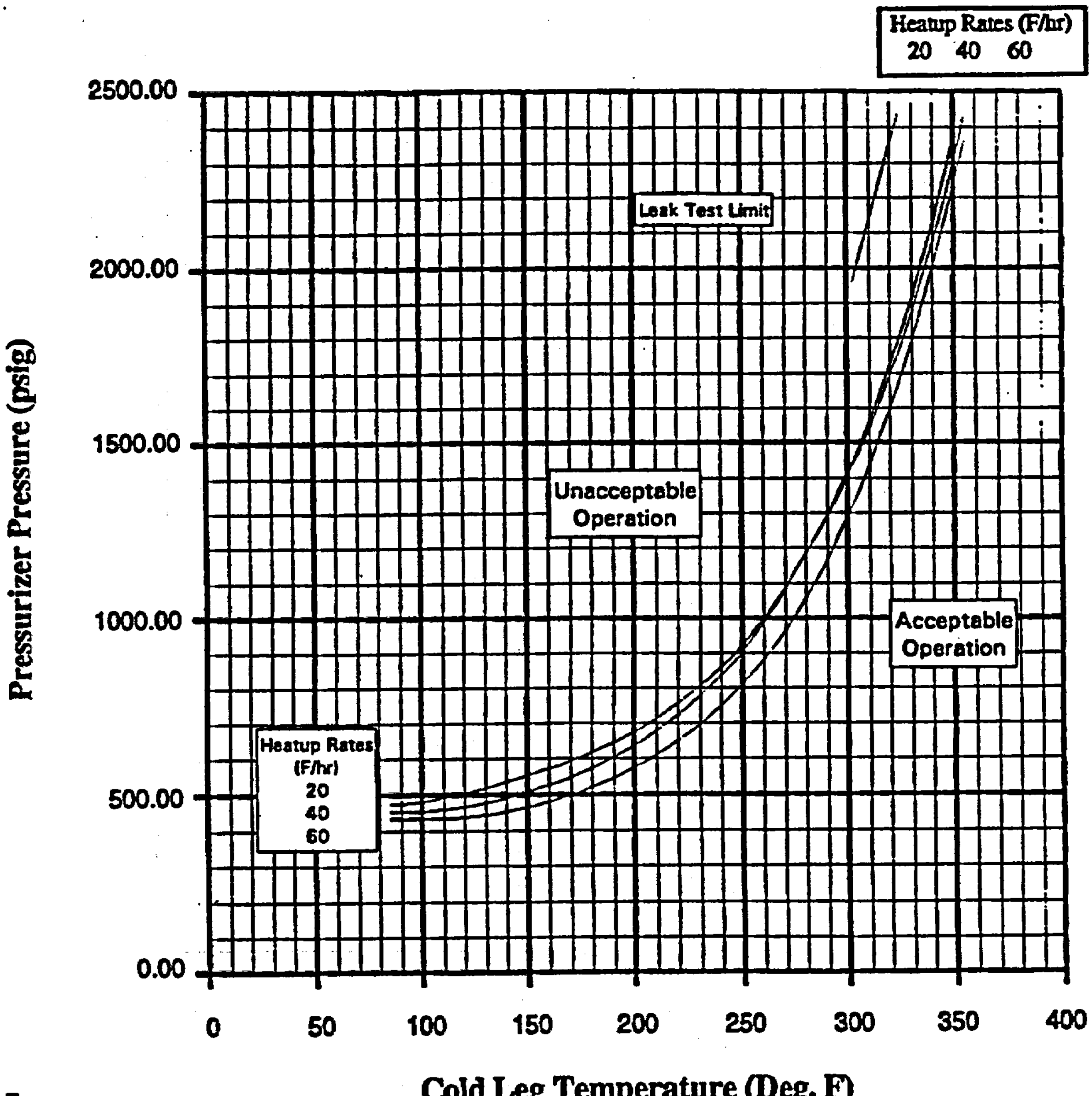
Summary of change:

This proposed change to the Technical Specifications is being made to extend the cumulative core burnup applicability limits for the P/T limits, LTOPS setpoints, and T_{enable} values.

DELETE	DATED	SUBSTITUTE
3/4 4-27	10-05-94	3/4 4-27
3/4 4-28	10-05-94	3/4 4-28
B 3/4 4-7	10-05-94	B 3/4 4-7
B 3/4 4-8	03-02-99	B 3/4 4-8

Figure 3.4-2 — North Anna Unit 2
Reactor Coolant System Heatup Limitations

Material Property Basis	
Limiting ART at 34.3 EFPY:	1/4-T, 218.5°F
	3/4-T, 195.6°F

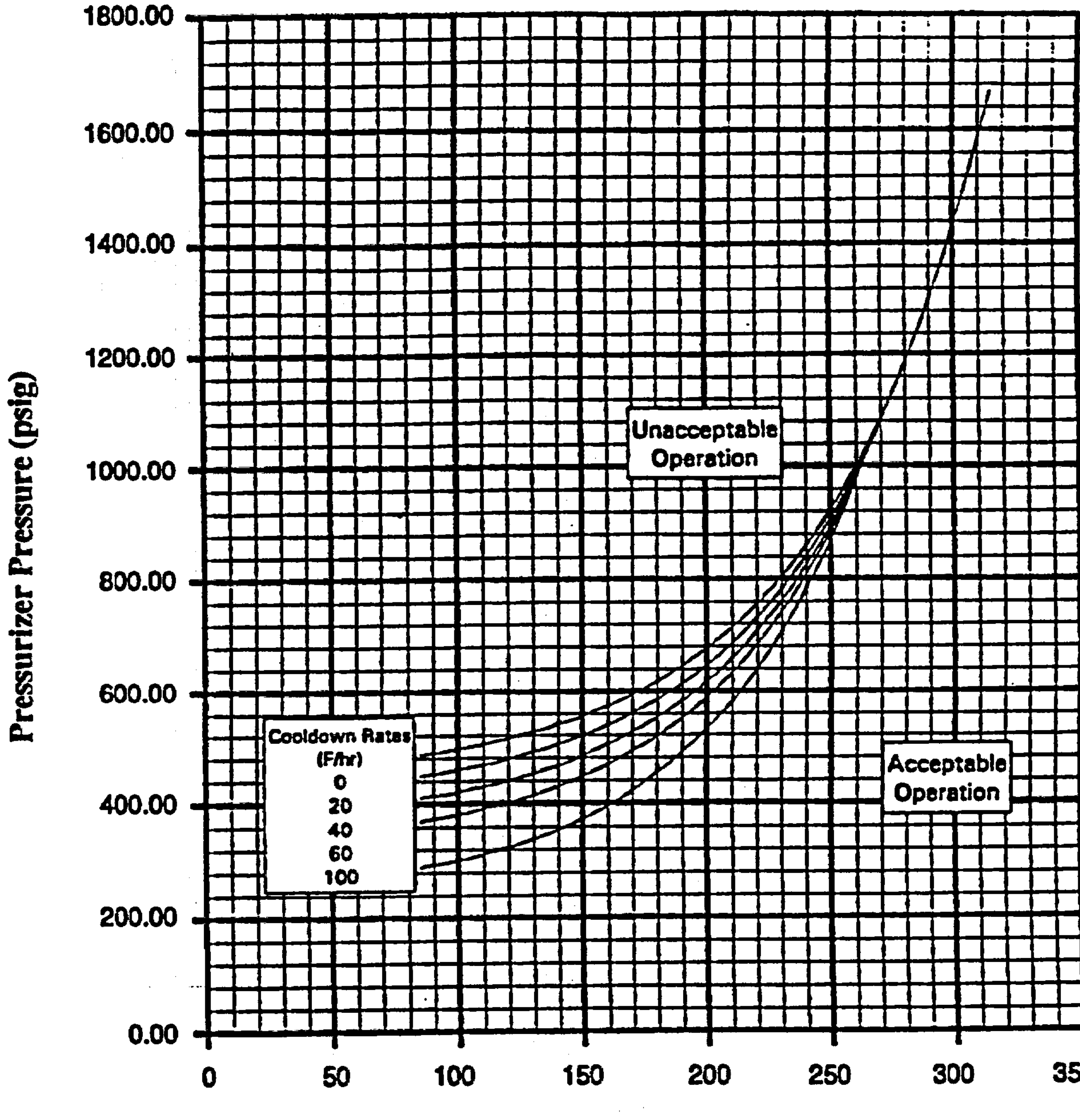


North Anna Unit 2 Reactor Coolant System Heatup Limitations (Heatup Rates up to 60 F/hr) Applicable for the First 34.3 EFPY (Without Margins for Instrumentation Errors)

N2CTS001

Figure 3.4-3 — North Anna Unit 2 Reactor Coolant System Cooldown Limitations

Material Property Basis	
Limiting ART at 34.3 EFPY:	1/4-T, 218.5°F
	3/4-T, 195.6°F



N2CTS002

North Anna Unit 2 Reactor Coolant System Cooldown Limitations (Cooldown Rates up to 100 F/hr) Applicable for the First 34.3 EFPY (Without Margins for Instrumentation Errors)

The heatup limit curve, Figure 3.4-2, is a composite curve which was prepared by determining the most conservative case, with either the inside or outside wall controlling, for any heatup rate up to 60°F per hour. The cooldown limit curves of Figure 3.4-3 are composite curves which were prepared based upon the same type analysis with the exception that the controlling location is always the inside wall where the cooldown thermal gradients tend to produce tensile stresses while producing compressive stresses at the outside wall. The heatup and cooldown curves of Figures 3.4-2 and 3.4-3 conservatively bound the design basis heatup and cooldown curves, which were based upon a 1/4-T RT_{NDT} value of 218.5°F and a 3/4-T RT_{NDT} value of 195.6°F. These RT_{NDT} values conservatively bound the predicted reactor vessel beltline RT_{NDT} values for North Anna Unit 2 operation through 34.3 EFPY.

The reactor vessel materials have been tested to determine their initial RT_{NDT}. Reactor operation and resultant fast neutron (E > 1 Mev) irradiation will cause an increase in the RT_{NDT}. An adjusted reference temperature, based upon the fluence and copper content of the material in question, can be predicted using US NRC Regulatory Guide 1.99, Revision 2. The heatup and cooldown limit curves (Figure 3.4-2 and Figure 3.4-3) include adjustments for this predicted shift in RT_{NDT} at the end of 34.3 EFPY.

The actual shift in the RT_{NDT} of the vessel material is established periodically by removal and evaluation of the reactor vessel material specimens installed on the inside wall of the thermal shield. The surveillance capsule withdrawal schedule was prepared in accordance with the requirements of ASTM E-185 and is presented in the UFSAR. Regulatory Guide 1.99, Revision 2, provides guidance for calculation of the shift in RT_{NDT} using measured data. Dosimetry from the surveillance capsule is used to provide benchmarks for calculation of the neutron fluence to which the material specimens and the reactor vessel were exposed.

The pressure-temperature limit lines shown on Figure 3.4-2 for inservice leak and hydrostatic testing have been provided to assure compliance with the minimum temperature requirements of Appendix G to 10 CFR 50. The minimum temperature for criticality specified in T.S. 3.1.1.5 assures compliance with the criticality limits of 10 CFR 50 Appendix G.

The number of reactor vessel irradiation surveillance specimens and the frequencies for removing and testing these specimens are provided in the UFSAR to assure compliance with the requirements of Appendix H to 10 CFR Part 50.

Pressurizer

The limitations imposed on pressurizer heatup and cooldown and spray water temperature differential are provided to assure that the pressurizer is operated within the design criteria assumed for the fatigue analysis performed in accordance with the ASME Code requirements.

Low-Temperature Overpressure Protection

The OPERABILITY of two PORVs or an RCS vent opening of greater than 2.07 square inches ensures that the RCS will be protected from pressure transients which could exceed the limits of Appendix G to 10 CFR Part 50 when one or more of the RCS cold legs are less than or equal to 270°F. Either PORV has adequate relieving capability to protect the RCS from overpressurization when the transient is limited to either (1) the start of an idle RCP with the secondary water temperature of the steam generator less than or equal to 50°F above the RCS cold leg temperatures or (2) the start of a charging pump and its injection into a water-solid RCS. The low temperature PORV lift setpoints were established to ensure that pressure at the reactor vessel beltline during these design basis events will not exceed 100% of the 10 CFR 50 Appendix G isothermal limit curve when the LTOP system is enabled.

Automatic or passive low temperature overpressure protection (LTOP) is required whenever any RCS cold leg temperature is less than 270°F. This temperature conservatively bounds the water temperature corresponding to a metal temperature of the limiting RT_{NDT}+31.9°F + instrument uncertainty. Above 270°F administrative control is adequate protection to ensure the limits of the heatup curve (Figure 3.4-2) and the cooldown curve (Figure 3.4-3) are not violated. The concept of requiring automatic LTOP at the lower end, and administrative control at the upper end, of the Appendix G curves is further discussed in NRC Generic Letter 88-11.

Surveillance limits are established for the pressure in the backup nitrogen accumulators to ensure there is adequate motive power for the PORVs to cope with an inadvertent start of a high head safety injection pump in a water solid condition, allowing adequate time for the operators to respond to terminate the event.

3/4.4.10 STRUCTURAL INTEGRITY

3/4.4.10.1 ASME CODE CLASS 1, 2 and 3 COMPONENTS

The inspection programs for ASME Code Class 1, 2 and 3 Reactor Coolant System components ensure that the structural integrity of these components will be maintained at an acceptable level throughout the life of the plant. To the extent applicable, the inspection program for components is in compliance with Section XI of the ASME Boiler and Pressure Vessel Code.

Attachment 4

Significant Hazards Consideration Determination

North Anna Power Station
Units 1 and 2
Virginia Electric and Power Company

SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Virginia Electric and Power Company has reviewed the requirements of 10 CFR 50.92 as they relate to the proposed changes for the North Anna Units 1 and 2 and determined that a significant hazards consideration is not involved. The proposed amendments to the North Anna Units 1 and 2 Technical Specifications extend the cumulative core burnup applicability limits for the Reactor Coolant System pressure/temperature (P/T) operating limits, Low Temperature Overpressure Protection System (LTOPS) setpoints, and LTOPS enable temperature (T_{enable}) values.

The proposed extension of the cumulative core burnup applicability limits is accomplished by revising the design basis P/T limit curves. The proposed revised design basis P/T limit curves utilize ASME Section XI Code Case N-640, which supports use of a conservative but less restrictive stress intensity formulation (K_{1c}). Therefore, the proposed revised design basis P/T limit curves are significantly less limiting than the existing Technical Specification P/T limit curves. The existing Technical Specification P/T limit curves and LTOPS setpoints have been demonstrated to remain conservative for the proposed extended cumulative core burnup applicability limit, and need not be changed. The following is provided to support this conclusion that the proposed changes do not create a significant hazards consideration.

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated

   The proposed changes extend the cumulative core burnup applicability of the existing North Anna Units 1 and 2 P/T limits, LTOPS setpoints, and T_{enable} values. No changes to plant systems, structures, or components are proposed, and no new allowable operating modes are established. The P/T limits, LTOPS setpoints, and T_{enable} values do not contribute to the probability of occurrence or consequences of accidents previously analyzed. The revised licensing basis analyses utilize acceptable analytical methods, and continue to demonstrate that established accident analysis acceptance criteria are met. Therefore, there is no increase in the probability or consequences of any accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated

   The proposed changes extend the cumulative core burnup applicability of the existing North Anna Units 1 and 2 P/T limits, LTOPS setpoints, and T_{enable} values. No changes to plant systems, structures, or components are proposed, and no new allowable operating modes are established. Therefore, the proposed changes do not create the possibility of any accident or malfunction of a different type previously evaluated.

3. Does the change involve a significant reduction in the margin of safety

   The proposed revised analysis bases use the ASME Section XI Code Case N-640 K_{1c} stress intensity formulation and a plant specific application of the analysis methodology which supports ASME Section XI Code Case N-514. These analysis features are less restrictive than those associated with the existing analyses, but are conservative with respect to established by ASME Section XI margins. The proposed revised analyses support continued use of the existing North Anna Units 1 and 2 Technical Specification P/T limit curves, LTOPS setpoints, LTOPS enable temperatures for North Anna Units 1 and 2 cumulative core burnups up to 32.3 effective full power years (EFPY) and 34.3 EFPY, respectively. The analyses demonstrate that established analysis acceptance criteria continue to be met. Specifically, the existing P/T limit curves, LTOPS setpoints, and LTOPS T_{enable} values provide acceptable margin to vessel fracture under both normal operation and LTOPS design basis (mass addition and heat addition) accident conditions. Therefore, the proposed changes do not result in a significant reduction in a margin of safety.

bc: Mr. J. R. Hayes - NAPS (w/o Att. 5)
Mr. T. B. Sowers - SPS
Mr. J. W. Crossman - NAPS
Mr. B. L. Stanley - SPS
Mr. D. A. Sommers - IN2SE
Mr. W. M. Oppenhimer – ~~IN3W~~ 3SW
Mr. E. T. Shaub - IN2SE
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Licensing File - GOV 02-54B
MSRC/NOB Coordinator - IN2SE
Records Management - GOV 02-54B (bc original) - IN-GW

CONCURRENCE

J. H. McCarthy ___SPS for JHM 6/19/00___

D. A. Sommers ___DAA 6/15/00___

D. A. Heacock ___DAH per telcon – CSS 6/17/00___

VERIFICATION OF ACCURACY

1. Approved North Anna Technical Specification Change Request No. 376, "Pressure/Temperature Operating Limits" dated June 7, 2000

2. Engineering Management and Vice President – Nuclear Operations present at the MSRC meeting on June 7, 2000 when Technical Specifications changes were approved

3. Engineering Transmittal NAF 2000-0031, Revision 0, "Revised Analysis for Reactor Coolant System Pressure/Temperature Operating Limits. LTOP Setpoints, and LTOPS Enable Temperatures" dated March 28, 2000

Required Changes to the UFSAR or the Topical Report:

1. Yes, UFSAR Change Request FN 2000-016

Action Plan/Commitments (Stated or Implied):

1. The required actions to implement these changes will be performed in accordance with the Technical Specification Change Request No. 376 Implementation Plan.

Attachment 5

Technical Report BAW-2356, Revision 1

BAW-2356, Revision 1
November 1999

Analysis of Capsule W
Virginia Power
North Anna Unit No. 1 Nuclear Power Plant

Reactor Vessel Material Surveillance Program

FRAMATOME
TECHNOLOGIES

BAW-2356, Revision 1
November 1999

Analysis of Capsule W
Virginia Power
North Anna Unit No. 1 Nuclear Power Plant

-- Reactor Vessel Material Surveillance Program --

by

M. J. DeVan
E. Giavedoni

FTI Document No. 77-2356-01
(See Section 9 for document signatures.)

Prepared for

Virginia Power

Prepared by

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FRAMATOME TECHNOLOGIES

Executive Summary

This report describes the results of the examination of the third capsule (Capsule W) of the Virginia Power North Anna Unit No. 1 as part of their reactor vessel surveillance program (RVSP). The objective of the program is to monitor the effects of neutron irradiation on the mechanical properties of the reactor vessel materials by testing and evaluation of tension test and Charpy V-notch impact specimens. The North Anna unit No. 1 RVSP was designed and furnished by Westinghouse Electric Corporation and was based on ASTM Standard E 185-73.

Capsule W was removed from the North Anna Unit No. 1 reactor vessel at the end-of-cycle 13 (EOC-13) for testing and evaluation. The capsule received an average fast fluence of 2.052 x 10^{19} n/cm^2 (E > 1.0 MeV). Based on the calculated cycle 11, 12, and 13 full power flux weighted average, the projected end-of-life (32.2 EFPY) peak fast fluence of the North Anna Unit No. 1 reactor vessel beltline region is 4.108 x 10^{19} n/cm^2 (E > 1.0 MeV).

The results of the tension tests indicated that the North Anna Unit No. 1 surveillance materials exhibited normal behavior relative to the neutron fluence exposure. The Charpy impact data results for the North Anna Unit No. 1 surveillance materials exhibited the characteristic behavior of transition temperature shifting to a higher temperature as a result of neutron fluence damage and a decrease in upper-shelf energy.

In accordance with Code of Federal Regulations, Title 10, Part 50.61, (10 CFR 50.61), the North Anna Unit No. 1 reactor vessel beltline materials will not exceed the PTS screening criteria before end-of-life (32.2 EFPY).

Acknowledgement

The author acknowledges the efforts of Kevin Hour of the B&W Services, Inc. Lynchburg Technology Center. His expertise in specimen testing contributed greatly to the success of this project.

Record of Revisions

Date	Revision No.	Description
September 1999	0	Original Issue
November 1999	1	Section 6 - Corrected text for Section 6.1. Section 9 - New signatures added.

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FRAMATOME TECHNOLOGIES

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1. Introduction

This report presents the examination results of the third reactor vessel surveillance capsule (Capsule W) removed from the Virginia Power's North Anna Unit No. 1 reactor vessel. The capsule was removed and the contents evaluated after being irradiated in the North Anna Unit No. 1 reactor as part of the reactor vessel surveillance program (RVSP) as documented in WCAP-8771.[1] This report describes the testing and the post-irradiation data obtained from the North Anna Unit No. 1 Capsule W after receiving an average fluence of 2.052×10^{19} n/cm^2 (E > 1.0 MeV). The data are compared to previous North Anna Unit No. 1 RVSP results from Capsule V[2] and Capsule U.[3]

The objective of the program is to monitor the effects of neutron irradiation on the mechanical properties of reactor vessel materials under actual plant operating conditions. The program was planned to monitor the effects of neutron irradiation on the reactor vessel materials for the 40-year design life of the reactor pressure vessel. The North Anna Unit No. 1 RVSP was designed and furnished by Westinghouse Electric Corporation and was based on American Society for Testing and Materials (ASTM) Standard E 185-73.[4]

2. Background

The ability of the reactor vessel to resist fracture is a primary factor in ensuring the safety of the primary system in light water-cooled reactors. The reactor vessel beltline region is the most critical region of the vessel because it is exposed to the highest level of neutron irradiation. The general effects of fast neutron irradiation on the mechanical properties of low-alloy ferritic steels used in the fabrication of reactor vessels are well characterized and documented. The low-alloy ferritic steels used in the beltline region of reactor vessels exhibit an increase in ultimate and yield strength properties with a corresponding decrease in ductility after irradiation. The most significant mechanical property change in reactor vessel steels is the increase in the ductile-to-brittle transition temperature accompanied by a reduction in the Charpy upper-shelf energy (C_vUSE) value.

Code of Federal Regulation, Title 10, Part 50, (10 CFR 50) Appendix G, *"Fracture Toughness Requirements,"* [5] specifies minimum fracture toughness requirements for the ferritic materials of the pressure-retaining components of the reactor coolant pressure boundary (RCPB) of light water-cooled power reactors and provides specific guidelines for determining the pressure-temperature limitations for operation of the RCPB. The fracture toughness and operational requirements are specified to provide adequate safety margins during any condition of normal operation, including anticipated operational occurrences and system hydrostatic tests, to which the pressure boundary may be subjected over its service lifetime. Although the requirements of 10 CFR 50, Appendix G, became effective on August 16, 1973, the requirements are applicable to all boiling and pressurized water-cooled nuclear power reactors, including those under construction or in operation on the effective date.

10 CFR 50, Appendix H, *"Reactor Vessel Materials Surveillance Program Requirements,"* [6] defines the material surveillance program required to monitor changes in the fracture toughness properties of ferritic materials in the reactor vessel beltline region of water-cooled reactors resulting from exposure to neutron irradiation and the thermal environment. Fracture toughness test data are obtained from material specimens contained in capsules that are periodically withdrawn from the reactor vessel. These data permit determination of the conditions under

which the vessel can be operated with adequate safety margins against non-ductile fracture throughout its service life.

A method for guarding against non-ductile fracture in reactor vessels is described in Appendix G to the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code, Section III, *"Nuclear Power Plant Components"*[7] and Section XI, *"Rules for Inservice Inspection."*[8] This method uses fracture mechanics concepts and the reference nil-ductility temperature, RT_{NDT}, which is defined as the greater of the drop weight nil-ductility transition temperature (in accordance with ASTM E 208-81[9]) or the temperature that is 60°F below that at which the material exhibits 50 ft-lbs and 35 mils lateral expansion. The RT_{NDT} of a given material is used to index that material to a reference stress intensity factor curve (K_{IR} curve), which appears in Appendix G of ASME B&PV Code Section III and Section XI. The K_{IR} curve is a lower bound of dynamic and crack arrest fracture toughness data obtained from several heats of pressure vessel steel. When a given material is indexed to the K_{IR} curve, allowable stress intensity factors can be obtained for the material as a function of temperature. The operating limits can then be determined using these allowable stress intensity factors.

The RT_{NDT} and, in turn, the operating limits of a nuclear power plant, are adjusted to account for the effects of irradiation on the fracture toughness of the reactor vessel materials. The irradiation embrittlement and the resultant changes in mechanical properties of a given pressure vessel steel can be monitored by a surveillance program in which surveillance capsules containing prepared specimens of the reactor vessel materials are periodically removed from the operating nuclear reactor and the specimens are tested. The increase in the Charpy V-notch 30 ft-lb temperature is added to the original RT_{NDT} to adjust it for irradiation embrittlement. The adjusted RT_{NDT} is used to index the material to the K_{IR} curve which, in turn, is used to set operating limits for the nuclear power plant. These new limits take into account the effects of irradiation on the reactor vessel materials.

10 CFR 50, Appendix G, also requires a minimum initial C_vUSE of 75 ft-lbs for all beltline region materials unless it is demonstrated that lower values of upper-shelf fracture energy will provide an adequate margin of safety against fracture equivalent to those required by ASME Section XI, Appendix G. No action is required for a material that does not meet the initial 75 ft-lbs requirement provided that the irradiation embrittlement does not cause the C_vUSE to drop below 50 ft-lbs. The regulations specify that if the C_vUSE drops below 50 ft-lbs, it must be demonstrated, in a manner approved by the Office of Nuclear Reactor Regulation, that the lower values will provide adequate margins of safety.

3. Surveillance Program Description

The reactor vessel surveillance program for North Anna Unit No. 1 includes eight capsules designed to monitor the effects of neutron and thermal environment on the materials of the reactor pressure vessel core region. The capsules, which were inserted into the reactor vessel before initial plant startup, were positioned inside the reactor vessel between the thermal shield and the vessel wall at the locations shown in Figure 3-1. WCAP-8771 includes a full description of the capsule locations and design. Capsule W was irradiated in the 245° position during the time of irradiation in the reactor vessel (cycles 1 through 13).

Capsule W was removed during the thirteenth refueling shutdown of the North Anna Unit No. 1 plant. The capsule contained Charpy V-notch (CVN) impact test specimens fabricated from one base metal forging (SA-508, Class 2), heat-affected-zone (HAZ) material, and a weld metal representative of the North Anna Unit No. 1 reactor vessel beltline region intermediate to lower shell circumferential weld. The tensile test specimens were fabricated from the same base metal forging and weld metal. In addition, wedge opening loading (WOL) specimens, fabricated from the base metal forging, were included in the capsule. The number of specimens of each material contained in Capsule W is described in Table 3-1, and the location of the individual specimens within the capsule is shown in Figure 3-2. The chemical compositions of the surveillance materials in Capsule W, obtained from the original surveillance program report,[1] are described in Table 3-2. In addition, a chemical analysis was performed on an irradiated Charpy base metal specimen (VT-71) and weld metal specimen (VW-71) from Capsule U.[3] The heat treatment of the surveillance materials in Capsule W is presented in Table 3-3.

All base metal specimens were machined from the ¼-thickness (¼T) location of the forging material after stress relieving. The base metal, HAZ material, and weld metal specimens were oriented such that the longitudinal axis of the specimen was either parallel or perpendicular to the principal working direction of the forging.

Capsule W contained dosimeter wires of copper, iron, nickel, and aluminum-0.15 weight percent cobalt (cadmium-shielded and unshielded) and cadmium-shielded neptunium-237 (^{237}Np) and uranium-238 (^{238}U). The location of these dosimeters within Capsule W is shown in Figure 3-2.

Thermal monitors fabricated from two low-melting alloys were included in the capsule. The thermal monitors were sealed in Pyrex tubes and inserted in spacers located in Figure 3-2. The eutectic alloys and their melting points are listed below:

2.5% Ag, 97.5% Pb	Melting Point 579°F
1.75% Ag, 0.75% Sn, 97.5% Pb	Melting Point 590°F

Table 3-1. Test Specimens Contained in North Anna Unit No. 1 Capsule W

Material Description	Number of Test Specimens		
	Tension	CVN Impact	WOL
Base Metal Forging 03 (Heat No. 990400/292332)			
Tangential	--	8	--
Axial	2	12	4
HAZ Metal	--	12	--
Weld Metal (Wire Ht. 25531 / Flux Lot 1211)	2	12	--
Total	4	44	4

Table 3-2. Chemical Composition of North Anna Unit No. 1 Capsule W Surveillance Materials

Element	Chemical Composition, wt%				
	Base Metal Forging 03 Heat No. 990400/292332			Weld Metal (Wire Ht. 25531 / Flux Lot 1211)	
	Westinghouse Analysis[1]	Rotterdam Dockyard Check Analysis[1]	Irradiated Charpy Specimen VT-71[3]	Westinghouse Analysis[1]	Irradiated Charpy Specimen VW-71[3]
C	0.20	0.19	---	0.06	---
Mn	0.68	0.68	0.749	1.29	1.45
P	0.019	0.010	0.010	0.020	0.022
S	0.011	0.014	---	0.012	---
Si	0.26	0.22	---	0.35	---
Ni	0.79	0.80	0.893	0.11	0.152
Mo	0.61	0.63	0.671	0.49	0.537
Cr	0.30	0.30	0.379	0.025	0.057
Cu	0.16	0.15	0.158	0.086	0.124
Al	0.021	---	---	0.009	---
Co	0.020	---	0.021	0.006	0.02
V	0.037	0.02	0.031	0.001	0.006
Sn	0.017	---	---	0.003	---
N_2	0.015	---	---	0.015	---

Table 3-3. Heat Treatment of North Anna Unit No. 1 Capsule W Surveillance Materials

Material	Heat Treatment
Base Metal Forging 03 Heat No. 990400/292332	1616-1725°F for 2½ hrs., water quenched 1202-1292°F for 7½ hrs., furnace cooled 1130±25°F for 14¾ hrs., furnace cooled
Weld Metal (Wire Ht. 25531 / Flux Lot 1211)	1130±25°F for 10¾ hrs., furnace cooled

Figure 3-1. Reactor Vessel Cross Section Showing Original Locations of RVSP Capsules in North Anna Unit No. 1 Reactor Vessel

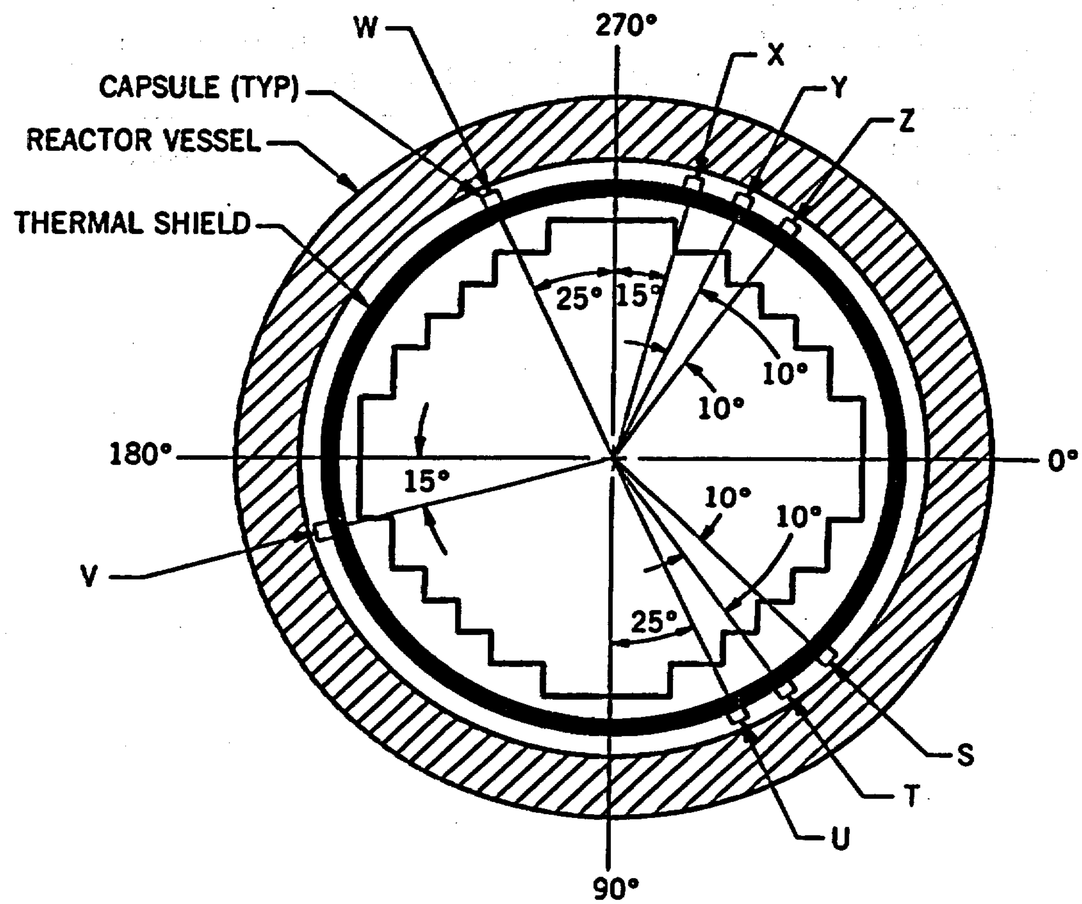
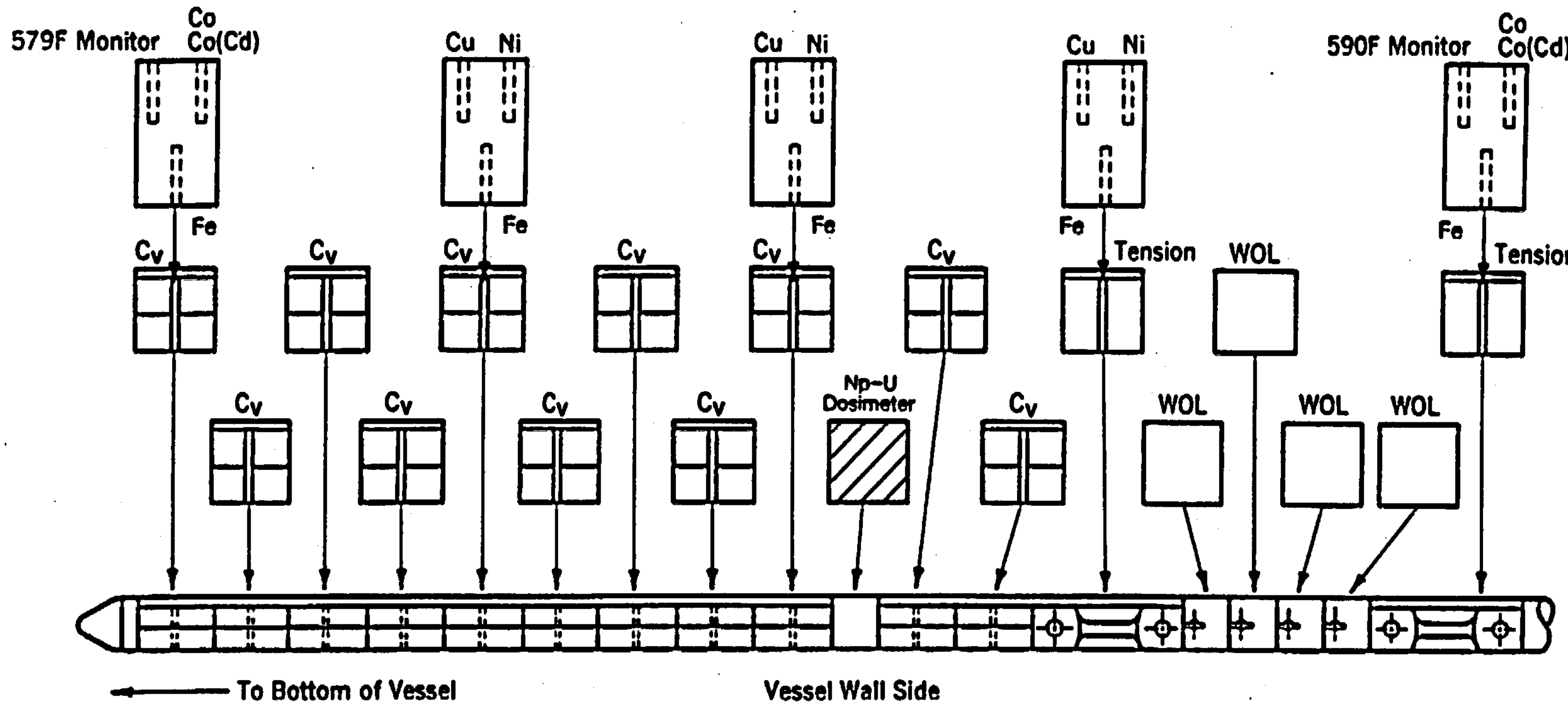


Figure 3-2. Surveillance Capsule Assembly Showing Locations of Specimens and Monitors for North Anna Unit No. 1 Capsule W



4. Tests of Unirradiated Material

Unirradiated material was evaluated for two purposes: (1) to establish baseline data to which irradiated properties data could be compared; and (2) to determine those material properties as required for compliance with 10 CFR 50, Appendices G and H.

Westinghouse Electric Corporation, as part of the development of the North Anna Unit No. 1 RVSP, performed the testing of the unirradiated surveillance material. The details of the testing procedures are described in Westinghouse Electric Corporation Report WCAP-8771. The unirradiated mechanical properties for the North Anna Unit No. 1 RVSP materials are summarized in Appendices C and D of this report.

The original unirradiated Charpy V-notch impact data were evaluated based on hand-fit Charpy curves generated using engineering judgment. These data were re-evaluated herein using a hyperbolic tangent curve-fitting program, and the results of the re-evaluation are presented in Appendix D. In addition, Appendix E contains a comparison of the Charpy V-notch shift results for each surveillance material, hand-fit versus hyperbolic tangent curve-fit.

5. Post-Irradiation Testing

The post-irradiation testing of the tension test specimens, the Charpy V-notch impact specimens, thermal monitors, and dosimeters for the North Anna Unit No. 1 Capsule W was performed at the BWX Technologies, Lynchburg Technology Center (LTC).[10]

5.1. Capsule Disassembly and Inventory

After capsule disassembly, the contents of Capsule W were inventoried and found to be consistent with the surveillance program report inventory (WCAP-8771). The capsule contained a total of 44 standard Charpy V-notch specimens, four (4) tensile specimens, four (4) WOL specimens, six (6) dosimetry blocks, and two (2) temperature monitors.

5.2. Thermal Monitors

The low-melting point (579°F and 590°F) eutectic alloys contained in Capsule W were x-rayed to reveal the shape of the monitors and examined for evidence of melting. No indication of melting was observed (see Figure 5-1). Therefore, based on this examination, the maximum temperature that the capsule test specimens were exposed to was less than 579°F.

5.3. Chemical Analysis Check Analysis

One tested irradiated base metal Charpy specimen and one tested irradiated weld metal Charpy specimen were analyzed to determine their chemical compositions. A small sample was removed from Specimen VT-36 (base metal) and from Specimen VW-29 (weld metal). Each sample was analyzed using the inductively coupled plasma (ICP) method to determine the following chemical constituents: manganese (Mn), phosphorous (P), sulfur (S), silicon (Si), nickel, (Ni), chromium (Cr), molybdenum (Mo), copper (Cu), cobalt (Co), vanadium (V). The results of the analyses are presented in Table 5-1.

5.4. Tension Test Results

The results of the post-irradiation tension test are presented in Table 5-2, and the stress-strain curves are presented in Figures 5-2 through 5-5. For the base metal Forging 03 material tests were performed at 300°F and 550°F, and for the weld metal material tests were performed at

200° and 550°F. The tests were performed using a MTS servohydraulic test machine. All tension tests were run using stroke control with an initial actuator travel rate of 0.0075 inch per minute. Following specimen yielding, an actuator speed of 0.03 inch per minute was used. The tension testing was performed in accordance with the applicable requirements of ASTM Standard E 21-92.[11] Photographs of the tension test specimen fractured surfaces are shown in Figures 5-6 and 5-7.

5.5. Charpy V-Notch Impact Test Results

The Charpy V-notch impact testing was performed in accordance with the applicable requirements of ASTM Standard E 23-91.[12] Impact energy, lateral expansion, and percent shear fracture were measured at numerous test temperatures and recorded for each specimen. The impact energy was measured using a certified Satec S1-1K Impact tester (traceable to NIST Standard) with a striker velocity of 16.90 ft/sec and 240 ft-lb of available energy. The lateral expansion was measured using a certified dial indicator. The specimen percent shear was estimated by video examination and comparison with the visual standards presented in ASTM Standard E 23-91. In addition, all Charpy V-notch impact testing was performed using instrumentation to record a load-versus-time trace and energy-versus-time trace for each impact event. The load-versus-time traces were analyzed to determine time, load, and impact energy for general yielding, maximum load, fast fracture, and crack arrest properties during the test. The dynamic yield stress is calculated from the three-point bend formula:

$$\sigma_y = 33.33 * (\textit{general yielding load})$$

The dynamic flow stress is calculated from the average of the yield and maximum loads, also using the three-point bend formula:

$$\sigma_{flow} = 33.33 * \left(\frac{(\textit{general yielding load} + \textit{maximum load})}{2} \right)$$

The results of the Charpy V-notch impact testing are shown in Tables 5-3 through 5-10 and Figures 5-8 through 5-11. The curves were generated using a hyperbolic tangent curve-fitting program to produce the best-fit curve through the data. The hyperbolic tangent (TANH) function (test response, i.e., absorbed energy, lateral expansion, and percent shear fracture, "R," as a function of test temperature, "T") used to evaluate the surveillance data is as follows:

$$R = A + B * \tanh\left[\frac{(T - To)}{C}\right]$$

The Charpy V-notch data was entered, and the coefficients *A, B, To,* and *C* are determined by the program minimizing the sum of the errors squared (least-squares fit) of the data points about the fitted curve. Using these coefficients and the above TANH function, a smooth curve is generated through the data for interpretation of the material transition region behavior. The coefficients determined for irradiated materials in Capsule W are shown in Table 5-11.

Photographs of the Charpy V-notch specimen fracture surfaces are presented in Figures 5-12 through 5-15.

5.6. Wedge Opening Loading Specimens

The wedge opening loading (WOL) specimens were not tested at the request of Virginia Power. The specimens are to be stored at the BWX Technologies LTC facility for possible future testing.

Table 5-1. Chemical Analysis Results of Selected Base Metal and Weld Metal Irradiated Charpy Specimens

Element	Chemical Composition, wt%	
	Irradiated Charpy Specimen VT-36	Irradiated Charpy Specimen VW-29
Mn	0.685	1.39
P	0.0631[a]	0.09
S	<0.0536[b]	0.04[a]
Si	0.248[a]	0.32[a]
Ni	0.785	0.11
Mo	0.701	0.57
Cr	0.323	0.03
Cu	0.155	0.0839
Co	0.0184	0.0188
V	0.0415	< 0.0016[b]

(a) Analyte present. Reported value is estimated; concentration is below the level for accurate qualification.

(b) Below minimum detection limit.

Table 5-2. Tensile Properties of North Anna Unit No. 1 Capsule W Reactor Vessel Surveillance Materials, Irradiated to 2.052 x 10^{19} n/cm^2 (E > 1.0 MeV)

Material	Specimen No.	Test Temp. (°F)	Strength		Fracture Properties			Elongation		Reduction in Area (%)
			Yield (ksi)	Ultimate (ksi)	Load (lb)	Stress (ksi)	Strength (ksi)	Uniform (%)	Total (%)	
Base Metal Forging 03 Heat No. 990400/292332 (Axial)	VT5	300	78.5	98.7	3706	142	75.5	7.31	14.4	47.0
	VT6	550	78.1	101.6	3908	114	79.6	8.57	13.1	30.4
Weld Metal (Wire Ht. 25531 / Flux Lot 1211)	VW6	200	73.0	86.8	3124	162	63.6	5.48	16.0	60.8
	VW5	550	72.9	89.2	3639	144	74.1	5.73	14.2	48.6

Table 5-3. Charpy V-Notch Impact Results for North Anna Unit No. 1 Capsule W Base Metal Forging 03, Heat No. 990400/292332, Irradiated to 2.052 x 10^{19} n/cm^2 (E>1.0 MeV) Tangential Orientation

Specimen ID	Test Temperature, °F	Impact Energy, ft-lbs	Lateral Expansion, mil	Shear Fracture, %
VL23	74	26.5	15	15
VL17	104	26.5	21	30
VL22	104	38.5	31	40
VL24	129	54.5	43	60
VL21	204	73.0	56	75
VL18	304	93.5*	76	100
VL20	354	93.5*	77	100
VL19	404	97.0*	83	100

* Value used to determine upper-shelf energy (USE) in accordance with ASTM Standard E 185-82.[13]

Table 5-4. Charpy V-Notch Impact Results for North Anna Unit No. 2 Capsule W Base Metal Forging 03, Heat No. 990400/292332, Irradiated to 2.052 x 10^{19} n/cm^2 (E > 1.0 MeV) Axial Orientation

Specimen ID	Test Temperature, °F	Impact Energy, ft-lbs	Lateral Expansion, mil	Shear Fracture, %
VT30	74	13.5	6	0
VT26	104	16.0	10	20
VT28	104	23.5	21	30
VT35	129	34.5	31	35
VT31	154	39.0	34	45
VT34	179	39.0	36	50
VT32	204	51.5	46	85
VT33	204	40.0	35	50
VT29	254	67.0	61	95
VT27	304	64.5*	60	100
VT36	354	69.5*	64	100
VT25	404	64.0*	65	100

*Value used to determine upper-shelf energy (USE) in accordance with ASTM Standard E 185-82.[13]

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Table 5-5. Charpy V-Notch Impact Results for North Anna Unit No. 1 Capsule W Weld Metal, Wire Heat 25531 / Flux Lot 1211, Irradiated to 2.052 x 10^{19} n/cm^2 (E>1.0 MeV)

Specimen ID	Test Temperature, °F	Impact Energy, ft-lbs	Lateral Expansion, mil	Shear Fracture, %
VW36	-76	8.5	7	0
VW27	-36	10.5	6	0
VW33	-36	30.0	26	0
VW30	4	14.0	14	10
VW29	44	31.5	29	45
VW26	74	23.5	22	45
VW34	74	44.5	40	55
VW35	104	42.5	42	55
VW31	129	61.5	55	75
VW32	204	81.5*	70	100
VW25	304	72.5*	71	100
VW28	404	69.0*	70	100

*Value used to determine upper-shelf energy (USE) in accordance with ASTM Standard E 185-82.[13]

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Table 5-6. Charpy V-Notch Impact Results for North Anna Unit No. 1 Capsule W Heat-Affected-Zone Material, Irradiated to 2.052 x 10^{19} n/cm^2 (E>1.0 MeV)

Specimen ID	Test Temperature, °F	Impact Energy, ft-lbs	Lateral Expansion, mil	Shear Fracture, %
VH28	-76	21.0	19	0
VH34	-36	8.0	5	5
VH29	4	30.5	15	10
VH27	44	56.0	37	60
VH25	74	91.0	55	N/A
VH31	74	10.5	5	10
VH26	104	36.0	30	65
VH36	104	36.5	32	55
VH30	129	82.5	58	65
VH33	204	92.5*	71	100
VH32	304	80.0*	64	100
VH35	404	95.5*	69	100

*Value used to determine upper-shelf energy (USE) in accordance with ASTM Standard E 185-82.[13]

Table 5-7. Instrumented Charpy V-Notch Properties of North Anna Unit No. 1 Capsule W, Base Metal Forging 03, Heat No. 990400/292332, Irradiated to 2.052×10^{19} n/cm^2 (E > 1.0 MeV) Tangential Orientation

Specimen ID	Test Temp. (F)	Charpy Energy (ft-lbf)	Yield Properties			Maximum Load Properties			Fast Fracture Properties			Crack Arrest Properties			Propagation Load Properties		Total Energy Properties		Yield Stress (ksi)	Flow Stress (ksi)
			Time (μsec)	Load (lbf)	Energy (ft-lbf)	Time (μsec)	Load (lbf)	Energy (ft-lbf)	Time (μsec)	Load (lbf)	Energy (ft-lbf)	Time (μsec)	Load (lbf)	Energy (ft-lbf)	Load (lbf)	Energy (ft-lbf)	Time (μsec)	Energy (ft-lbf)		
VL23	74	26.5	151	3793	5.4	395	4499	22.7	395	4499	22.7	453	0	25.2	4517	2.6	453	25.2	126.4	138.2
VL17	104	26.5	169	3606	5.7	387	4345	20.7	387	4345	20.7	452	0	23.2	4359	2.5	452	23.2	120.2	132.5
VL22	104	38.5	163	3646	5.0	539	4632	32.0	569	4575	34.4	640	0	37.3	4598	5.3	640	37.3	121.5	138.0
VL24	129	54.5	164	3531	4.8	622	4566	37.0	752	4340	46.8	872	789	50.4	3551	18.3	2860	55.3	117.7	134.9
VL21	204	73.0	186	3455	4.4	646	4460	35.4	1000	3834	60.4	1104	2001	65.2	1833	42.8	3078	78.2	115.2	131.9
VL18	304	93.5	164	3172	5.0	616	4170	34.5	N/A	N/A	N/A	N/A	N/A	N/A	0	66.8	3444	101.3	105.7	122.4
VL20	354	93.5	160	3112	4.8	612	4138	33.9	N/A	N/A	N/A	N/A	N/A	N/A	0	66.9	3406	100.9	103.7	120.8
VL19	404	97.0	166	2960	4.4	624	4039	32.4	N/A	N/A	N/A	N/A	N/A	N/A	0	70.8	3840	103.2	98.7	116.6

Table 5-8. Instrumented Charpy V-Notch Properties of North Anna Unit No. 1 Capsule W, Base Metal Forging 03, Heat No. 990400/292332, Irradiated to 2.052 x 10^{19} n/cm^2 (E > 1.0 MeV) Axial Orientation

Specimen ID	Test Temp. (F)	Charpy Energy (ft-lbf)	Yield Properties			Maximum Load Properties			Fast Fracture Properties			Crack Arrest Properties			Propagation Load Properties		Total Energy Properties		Yield Stress (ksi)	Flow Stress (ksi)
			Time (μsec)	Load (lbf)	Energy (ft-lbf)	Time (μsec)	Load (lbf)	Energy (ft-lbf)	Time (μsec)	Load (lbf)	Energy (ft-lbf)	Time (μsec)	Load (lbf)	Energy (ft-lbf)	Load (lbf)	Energy (ft-lbf)	Time (μsec)	Energy (ft-lbf)		
VT30	74	13.5	148	3687	5.0	218	3855	9.4	218	3855	9.4	270	0	11.3	3857	1.9	270	11.3	122.9	125.7
VT26	104	16.0	164	3609	4.9	255	3963	10.6	255	3963	10.6	364	637	13.1	3326	4.4	1170	15.0	120.3	126.2
VT28	104	23.5	156	3577	4.9	343	4214	17.1	360	4177	18.3	429	2.3	20.8	4175	3.7	429	20.8	119.2	129.8
VT35	129	34.5	164	3462	5.4	492	4283	27.6	497	4280	28	614	741	31.1	3540	6.0	1729	33.6	115.4	129.1
VT31	154	39.0	168	3411	5.3	533	4347	30.1	533	4347	30.1	661	902	34.5	3445	8.8	1766	38.9	113.7	129.3
VT34	179	39.0	170	3321	5.3	434	4145	22.4	434	4145	22.4	554	2011	27.4	2144	17.3	2814	39.7	110.7	124.4
VT32	204	51.5	168	3266	5.0	532	4039	28.5	536	4039	28.8	702	2742	38.3	1297	24.8	2656	53.4	108.9	121.7
VT33	204	40.0	172	3303	5.2	370	3892	17.5	370	3892	17.5	486	2277	22.8	1615	22.9	3060	40.4	110.1	119.9
VT29	254	67.0	138	3048	3.5	538	4158	28.9	894	3140	52	982	1645	55.3	1495	40.2	3018	69.0	101.6	120.1
VT27	304	64.5	158	3048	4.6	524	3871	27.0	N/A	N/A	N/A	N/A	N/A	N/A	0	39.8	2958	66.8	101.6	115.3
VT36	354	69.5	168	3045	5.0	530	3935	27.2	N/A	N/A	N/A	N/A	N/A	N/A	0	45.5	2920	72.7	101.5	116.3
VT25	404	64.0	164	2972	4.8	534	3864	27.0	N/A	N/A	N/A	N/A	N/A	N/A	0	39.4	3006	66.4	99.1	113.9

Table 5-9. Instrumented Charpy V-Notch Properties of North Anna Unit No. 1 Capsule W, Weld Metal, Wire Heat 25531 / Flux Lot 1211, Irradiated to 2.052 x 10^{19} n/cm^2 (E > 1.0 MeV)

Specimen ID	Test Temp. (F)	Charpy Energy (ft-lbf)	Yield Properties			Maximum Load Properties			Fast Fracture Properties			Crack Arrest Properties			Propagation Load Properties		Total Energy Properties		Yield Stress (ksi)	Flow Stress (ksi)
			Time (μsec)	Load (lbf)	Energy (ft-lbf)	Time (μsec)	Load (lbf)	Energy (ft-lbf)	Time (μsec)	Load (lbf)	Energy (ft-lbf)	Time (μsec)	Load (lbf)	Energy (ft-lbf)	Load (lbf)	Energy (ft-lbf)	Time (μsec)	Energy (ft-lbf)		
VW36	-76	8.5	133	3632	3.7	160	3767	5.4	160	3767	5.4	224	0	8.0	3795	2.6	224	8.0	121.1	123.3
VW27	-36	10.5	161	3650	5.2	208	3675	8.2	208	3675	8.2	262	16.1	10.2	3659	2.0	262	10.2	121.7	122.1
VW33	-36	30.0	155	3583	5.1	509	4140	28.7	509	4140	28.7	566	2.3	31.1	4154	2.4	566	31.1	119.4	128.7
VW30	4	14.0	154	3482	4.5	253	3664	10.4	253	3664	10.4	319	0	12.8	3662	2.5	319	12.8	116.1	119.1
VW29	44	31.5	141	3255	4.2	496	3853	25.8	496	3853	25.8	604	828	28.9	3025	5.4	1712	31.2	108.5	118.5
VW26	74	23.5	140	3186	4.1	244	3443	9.9	244	3443	9.9	381	2001	15.6	1442	13.8	1476	23.6	106.2	110.5
VW34	74	44.5	136	3080	4.0	697	3993	38.7	697	3993	38.7	843	885	44.1	3048	7.5	1383	46.2	102.7	117.9
VW35	104	42.5	148	2946	3.6	626	3813	32.0	668	3802	34.7	790	796	38.1	3006	11.1	2488	43.2	98.2	112.6
VW31	129	61.5	142	2834	3.5	708	3685	36.4	908	3494	48.6	1130	1240	55.9	2254	27.6	2760	64.0	94.5	108.6
VW32	204	81.5	142	2677	3.4	710	3632	35.4	N/A	N/A	N/A	N/A	N/A	N/A	0	51.5	3142	86.9	89.2	105.1
VW25	304	72.5	144	2544	3.4	620	3397	28.4	N/A	N/A	N/A	N/A	N/A	N/A	0	47.2	2760	75.5	84.8	99.0
VW28	404	69.0	146	2468	3.5	622	3257	27.6	N/A	N/A	N/A	N/A	N/A	N/A	0	43.4	2868	71.0	82.3	95.4

Table 5-10. Instrumented Charpy V-Notch Properties of North Anna Unit No. 1 Capsule W, Heat-Affect-Zone Material, Irradiated to 2.052 x 10^{19} n/cm^2 (E > 1.0 MeV)

Specimen ID	Test Temp. (F)	Charpy Energy (ft-lbf)	Yield Properties			Maximum Load Properties			Fast Fracture Properties			Crack Arrest Properties			Propagation Load Properties		Total Energy Properties		Yield Stress (ksi)	Flow Stress (ksi)
			Time (μsec)	Load (lbf)	Energy (ft-lbf)	Time (μsec)	Load (lbf)	Energy (ft-lbf)	Time (μsec)	Load (lbf)	Energy (ft-lbf)	Time (μsec)	Load (lbf)	Energy (ft-lbf)	Load (lbf)	Energy (ft-lbf)	Time (μsec)	Energy (ft-lbf)		
VH28	-76	21.0	153	4267	5.4	326	4717	18.4	326	4717	18.4	381	0	20.9	4745	2.5	381	20.9	142.2	149.7
VH34	-36	8.0	139	4041	4.5	139	4041	4.5	139	4041	4.5	189	0	6.5	4060	1.9	189	6.5	134.7	134.7
VH29	4	30.5	156	3986	5.3	439	4750	26.2	464	4724	28.2	523	0	31.0	4727	4.8	523	31.0	132.9	145.6
VH27	44	56.0	151	3751	5.1	700	4720	45.7	839	4584	56.7	898	0	59.4	4614	13.7	898	59.4	125.0	141.2
VH25	74	91.0	156	3710	5.6	606	4738	38.7	1343	2875	90.4	1443	1573	93.7	1302	60.7	2318	99.5	123.7	140.8
VH31	74	10.5	159	3689	5.5	159	3689	5.5	159	3689	5.5	235	0	8.3	3689	3.1	323	8.6	123.0	123.0
VH26	104	36.0	165	3567	5.6	437	4310	24.1	473	4290	26.7	584	1283	31	3006	12.2	1686	36.3	118.9	131.3
VH36	104	36.5	168	3623	4.9	448	4303	23.8	530	4262	29.7	644	706	33.2	3556	12.9	1754	36.8	120.8	132.1
VH30	129	82.5	166	3425	5.2	706	4497	43.1	1274	3015	80.5	1386	1392	83.9	1624	45.4	2370	88.5	114.2	132.0
VH33	204	92.5	174	3181	5.2	714	4306	41.1	N/A	N/A	N/A	N/A	N/A	N/A	0	58.6	2638	99.7	106.0	124.8
VH32	304	80.0	170	3034	4.9	620	3974	32.6	N/A	N/A	N/A	N/A	N/A	N/A	0	50.4	2908	83.0	101.1	116.8
VH35	404	95.5	160	3018	4.3	716	4073	39.0	N/A	N/A	N/A	N/A	N/A	N/A	0	63.5	2782	102.5	100.6	118.2

Table 5-11. Hyperbolic Tangent Curve Fit Coefficients for the North Anna Unit No. 1 Capsule W Surveillance Materials

Material Description	Hyperbolic Tangent Curve Fit Coefficients		
	Absorbed Energy	Lateral Expansion	Percent Shear Fracture
Base Metal Forging 03 Ht. No. 990400/292332 (Tangential)	A: 49.3 B: 47.1 C: 108.4 T0: 135.2	A: 41.0 B: 40.0 C: 117.0 T0: 147.3	A: 50.0 B: 50.0 C: 90.6 T0: 130.5
Base Metal Forging 03 Ht. No. 990400/292332 (Axial)	A: 35.5 B: 33.3 C: 110.1 T0: 154.2	A: 33.4 B: 32.4 C: 107.6 T0: 165.6	A: 50.0 B: 50.0 C: 89.4 T0: 165.9
Weld Metal (Wire Ht. 25531 / Flux Lot 1211)	A: 39.7 B: 37.5 C: 121.6 T0: 73.7	A: 37.6 B: 36.6 C: 128.3 T0: 80.6	A: 50.0 B: 50.0 C: 85.4 T0: 77.4
HAZ Metal	A: 48.4 B: 46.2 C: 156.3 T0: 74.3	A: 36.5 B: 35.5 C: 138.0 T0: 85.7	A: 50.0 B: 50.0 C: 98.9 T0: 91.4

Figure 5-1. Photographs of Thermal Monitors Removed from the North Anna Unit No. 1 RVSP Capsule W

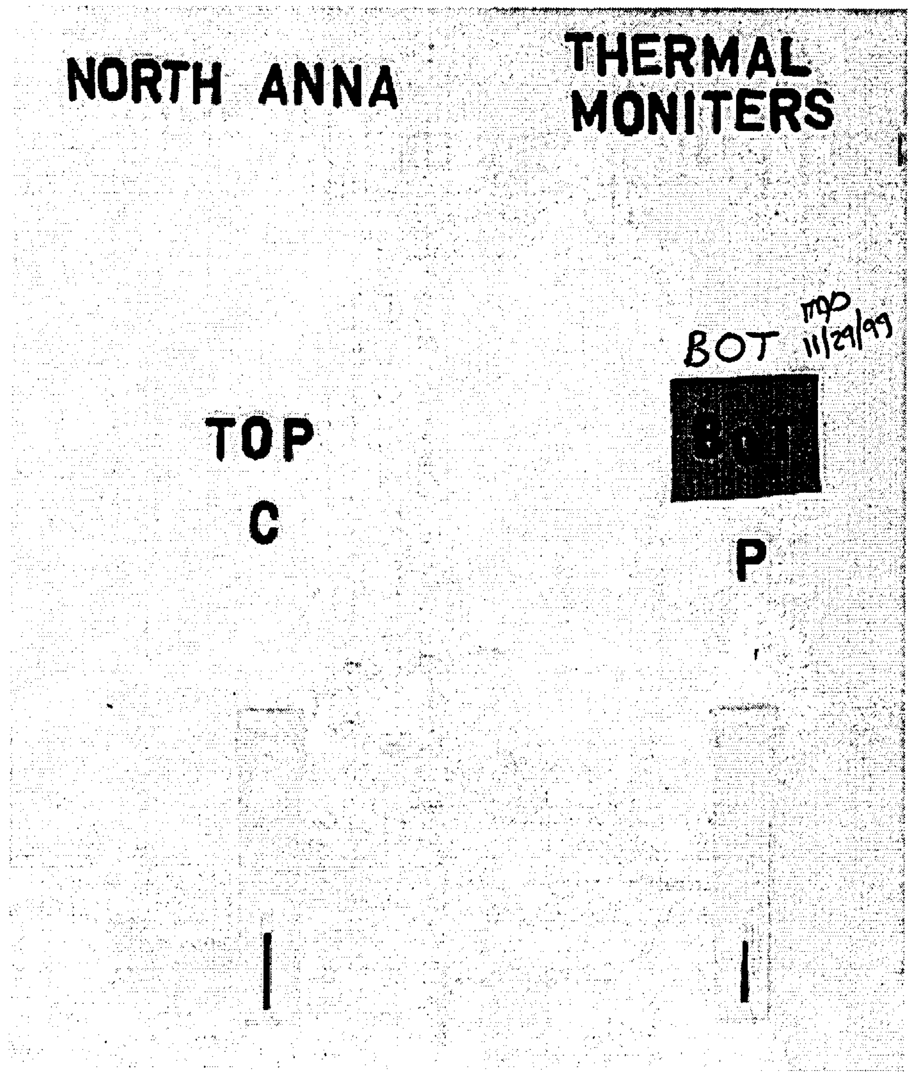

Figure 5-2. Tension Test Stress-Strain Curve for Base Metal Forging 03, Heat No. 990400/292332, Axial Orientation, Specimen No. VT5, Tested at 300°F

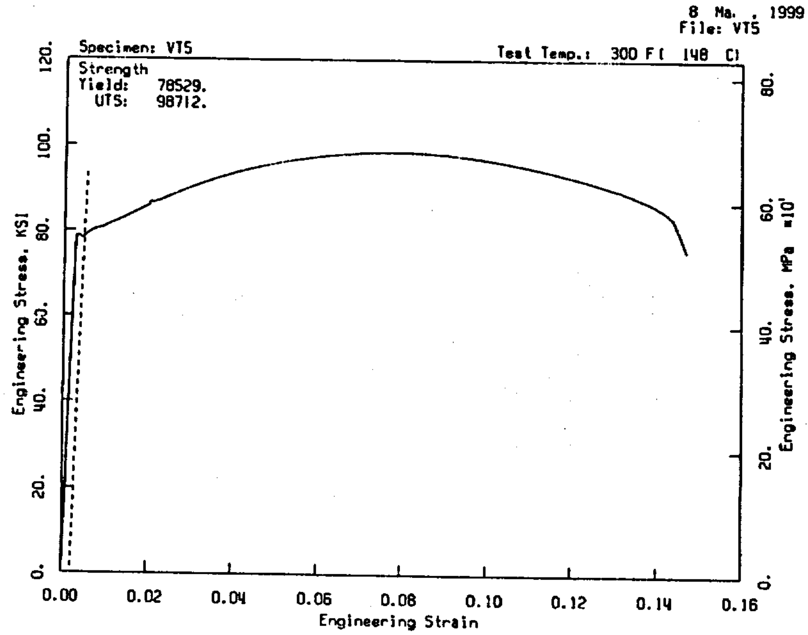

Figure 5-3. Tension Test Stress-Strain Curve for Base Metal Forging 03, Heat No. 990400/292332, Axial Orientation, Specimen No. VT6, Tested at 550°F

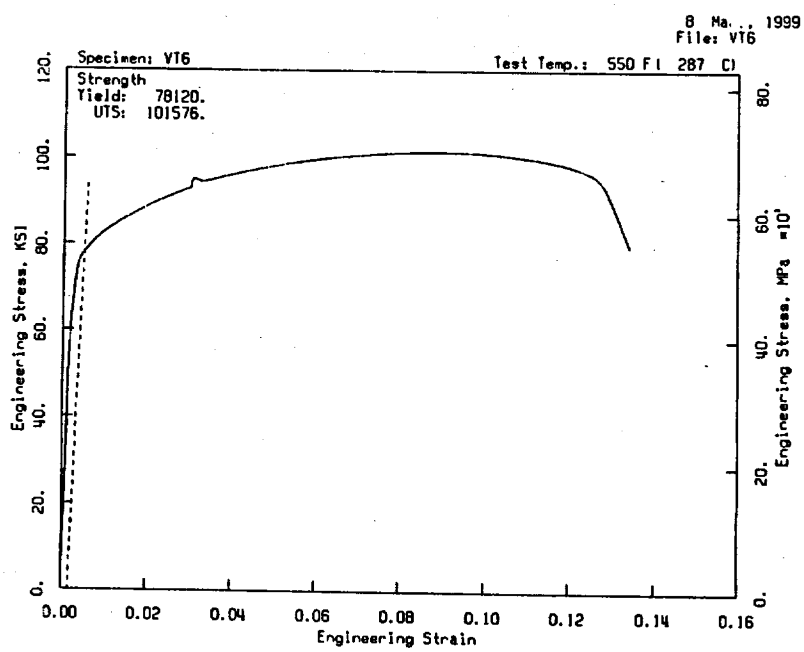

Figure 5-4. Tension Test Stress-Strain Curve for Weld Metal, Wire Heat 25531 / Flux Lot 1211, Specimen No. VW6, Tested at 200°F

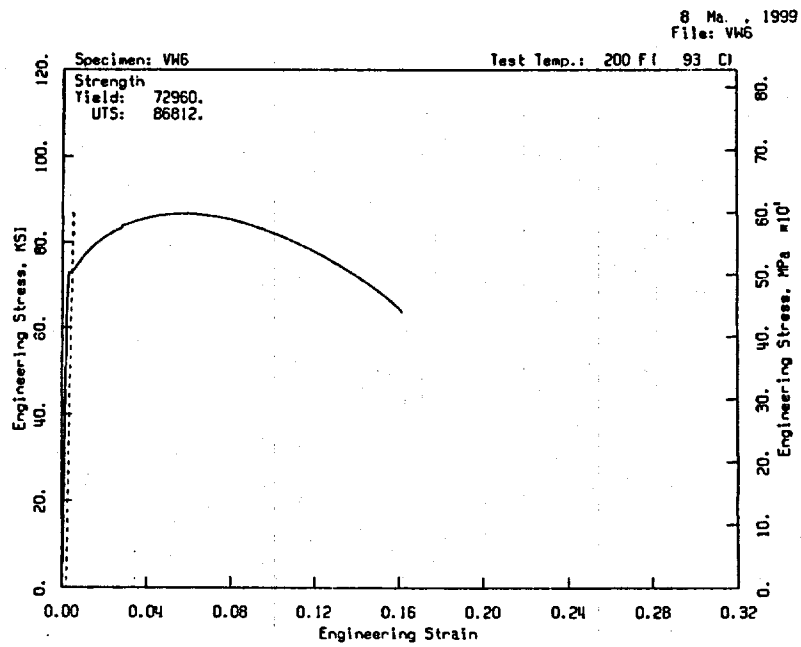

Figure 5-5. Tension Test Stress-Strain Curve for Weld Metal, Wire Heat 25531 / Flux Lot 1211, Specimen No. VW5, Tested at 550°F

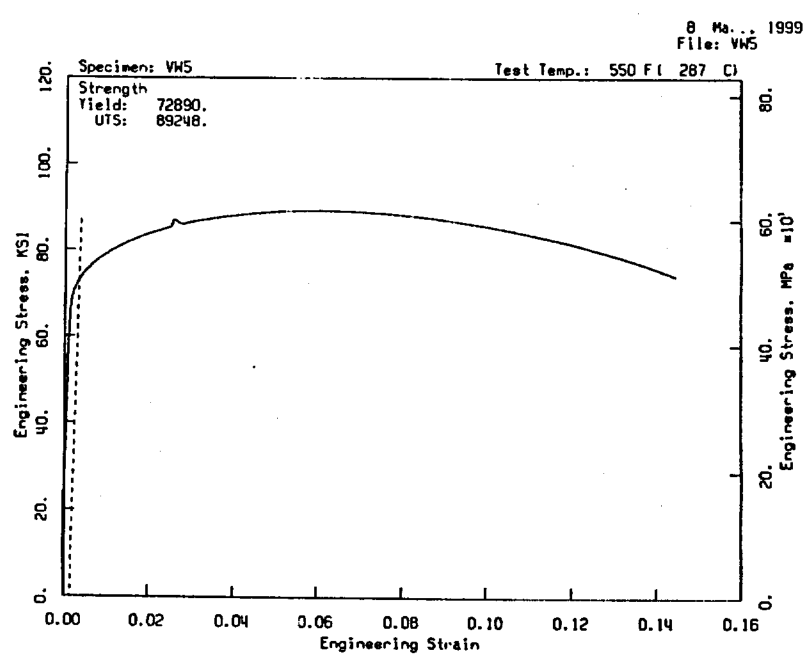

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Figure 5-6. Photographs of Tested Tension Test Specimens and Corresponding Fracture Surfaces – Base Metal Forging 03, Heat No. 990400/292332 Axial Orientation

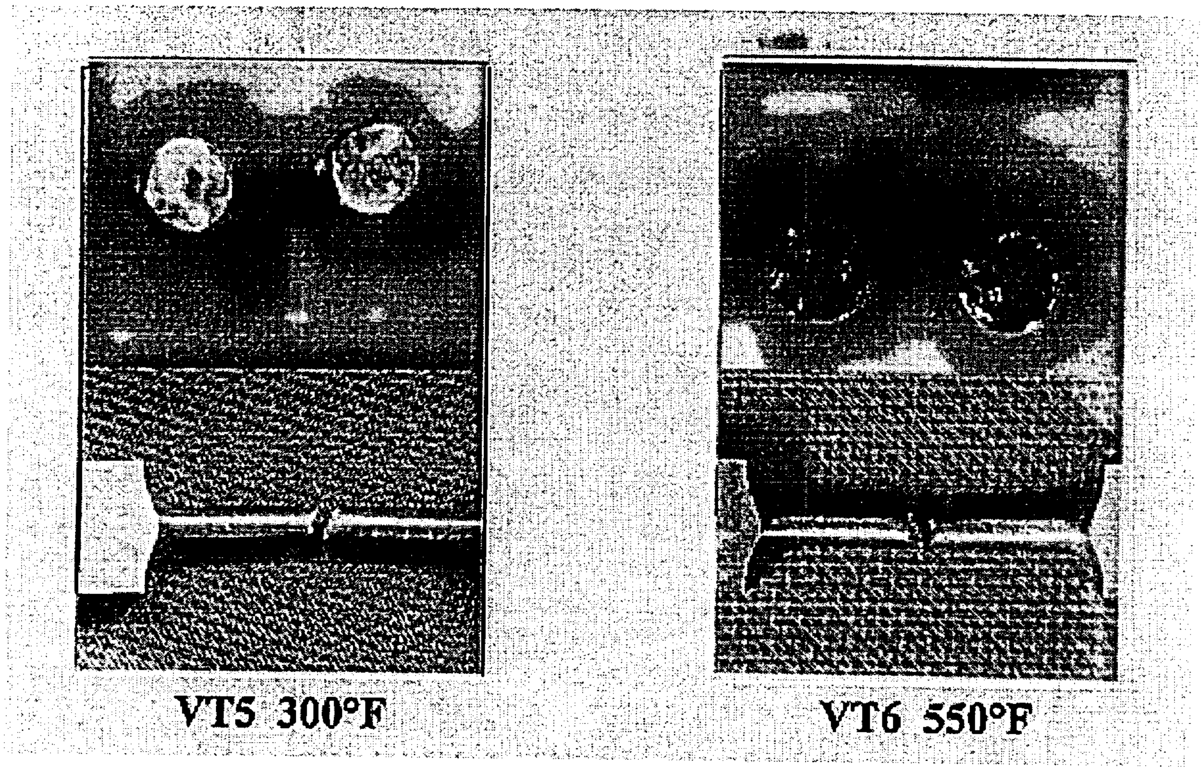
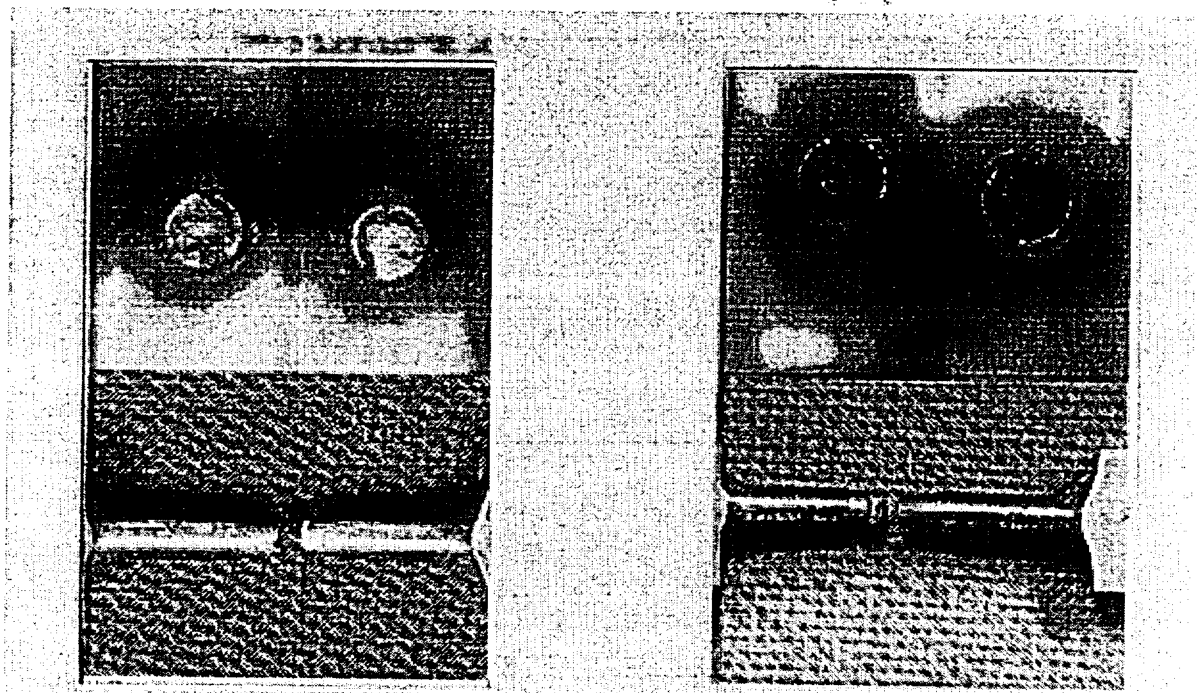


Figure 5-7. Photographs of Tested Tension Test Specimens and Corresponding Fracture Surfaces – Weld Metal, Wire Heat 25531 / Flux Lot 1211



VW6 200°F VW5 550°F

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Figure 5-8. Charpy Impact Data for Irradiated Base Metal Forging 03, Heat No. 990400/292332, Tangential Orientation

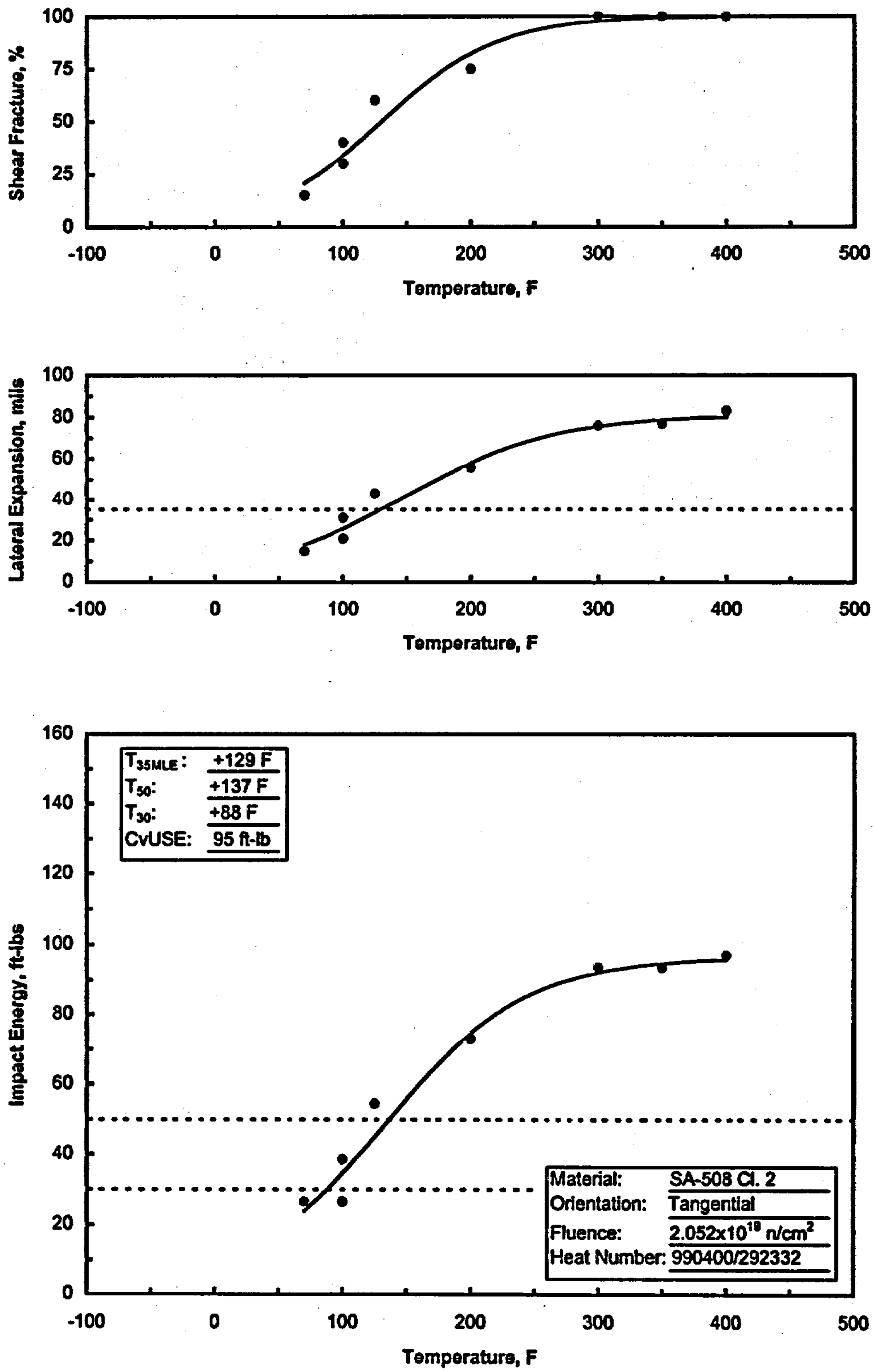

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Figure 5-9. Charpy Impact Data for Irradiated Base Metal Forging 03, Heat No. 990400/292332, Axial Orientation

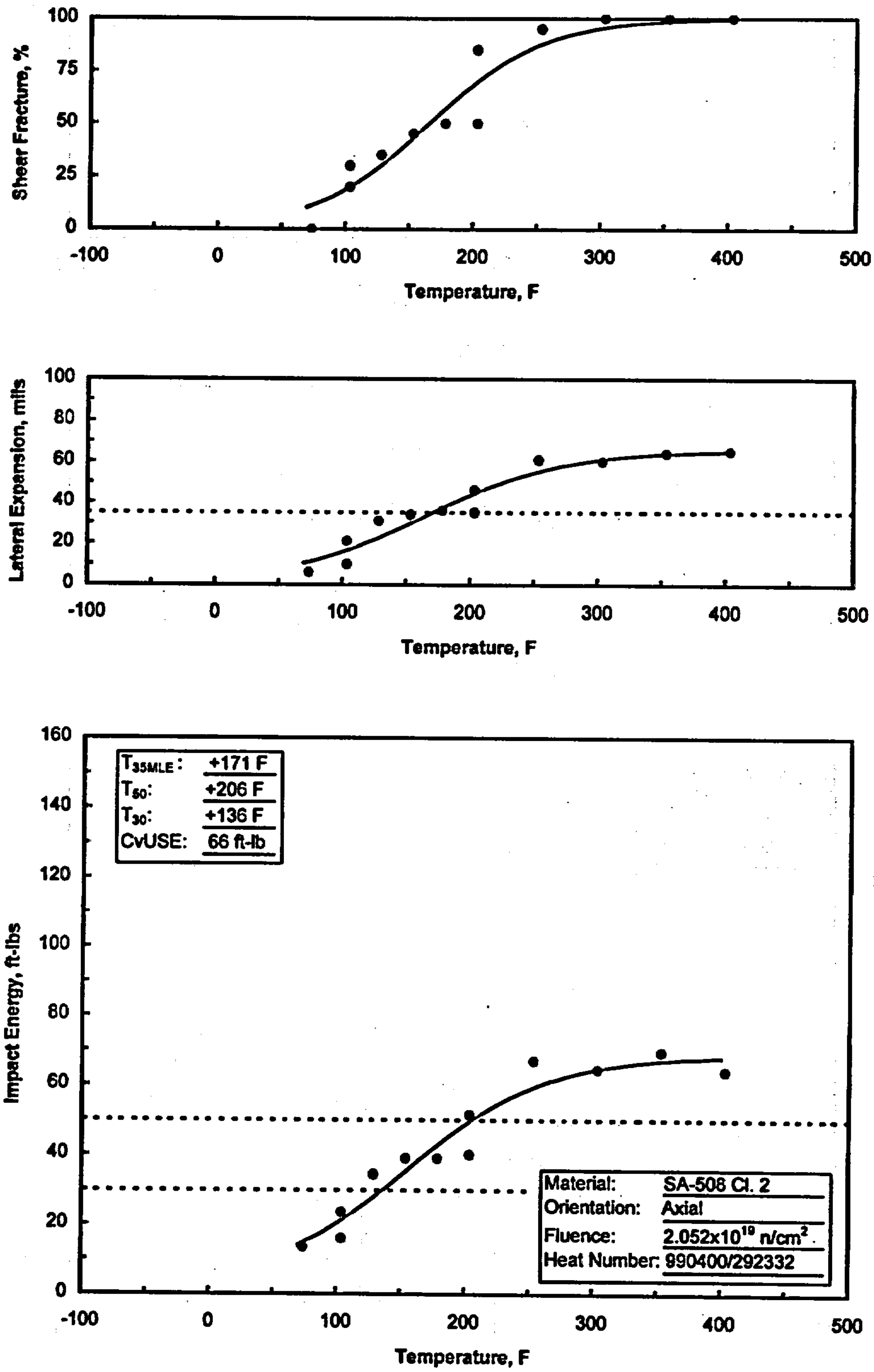

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Figure 5-10. Charpy Impact Data for Irradiated Weld Metal Wire Heat 25531 / Flux Lot 1211

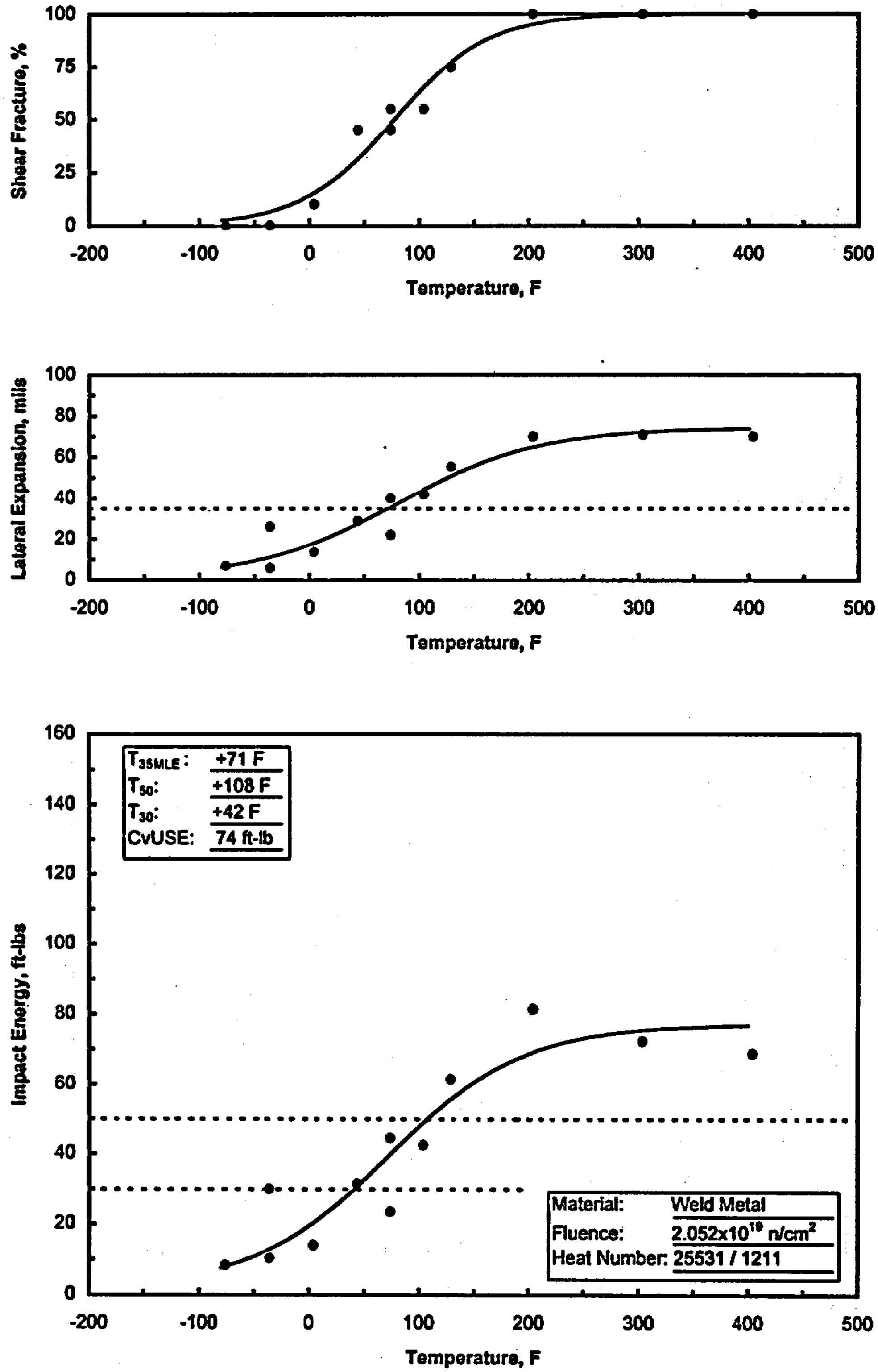

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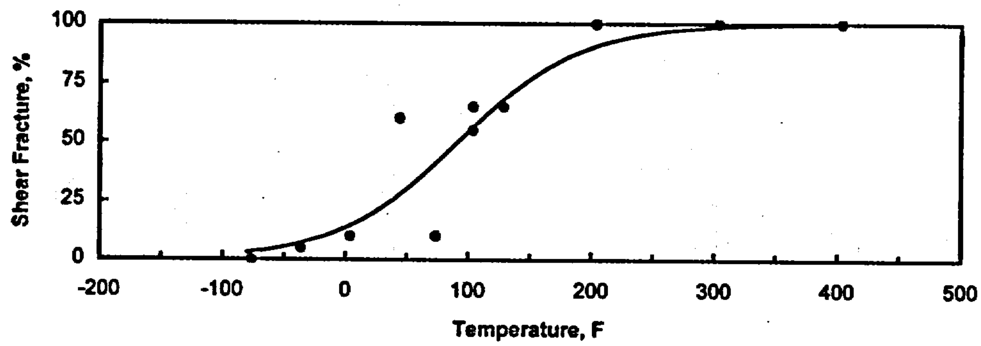
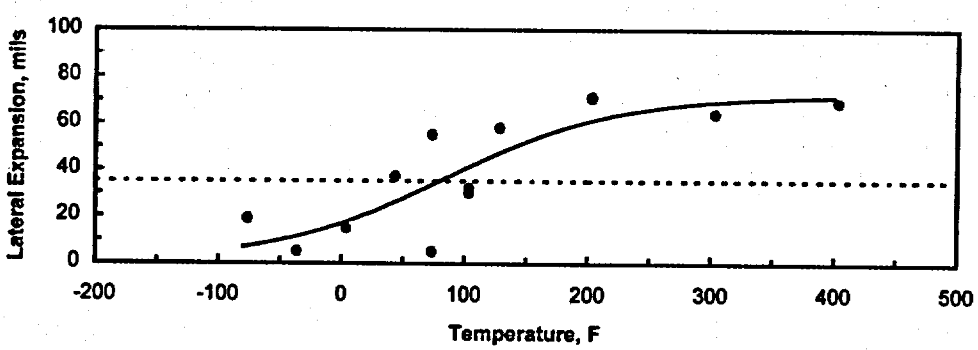
Figure 5-11. Charpy Impact Data Irradiated Heat-Affected-Zone Material





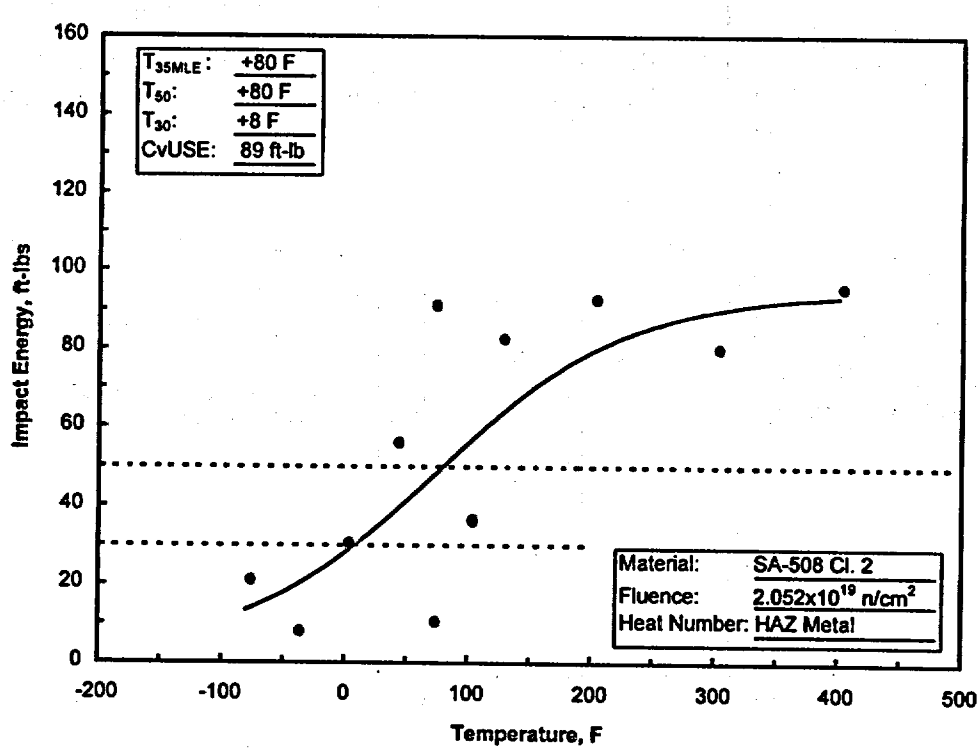


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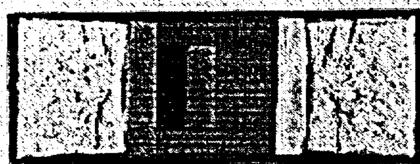
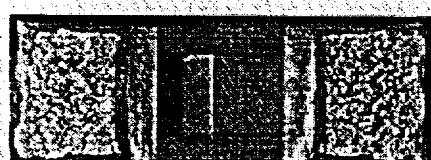
Figure 5-12. Photographs of Charpy Impact Specimen Fracture Surfaces, Base Metal Forging 03, Heat No. 990400/292332, Tangential Orientation

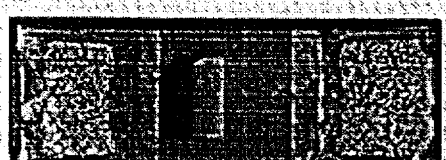


Specimen No. VL23, Test Temperature 74°F

Specimen No. VL17, Test Temperature 104°F



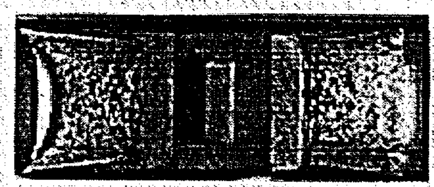
Specimen No. VL22, Test Temperature 104°F



Specimen No. VL24, Test Temperature 129°F

Specimen No. VL21, Test Temperature 204°F

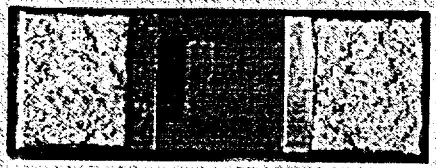
Specimen No. VL18, Test Temperature 304°F



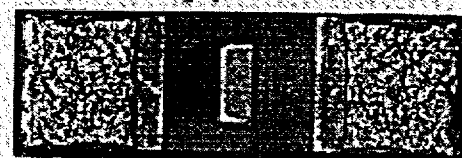
Specimen No. VL20, Test Temperature 354°F

Specimen No. VL19, Test Temperature 404°F

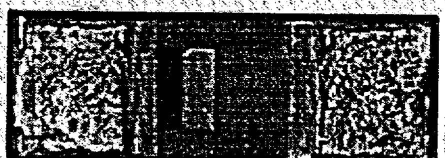
Figure 5-13. Photographs of Charpy Impact Specimen Fracture Surfaces, Base Metal Forging 03, Heat No. 990400/292332, Axial Orientation



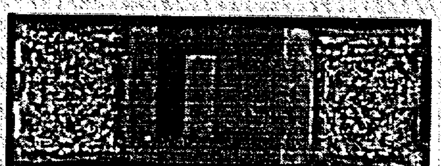
Specimen No. VT30, Test Temperature 74°F



Specimen No. VT26, Test Temperature 104°F

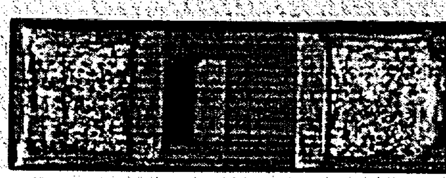


Specimen No. VT28, Test Temperature 104°F



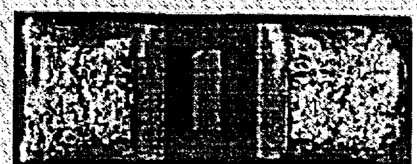
Specimen No. VT35, Test Temperature 129°F

Specimen No. VT31, Test Temperature 154°F

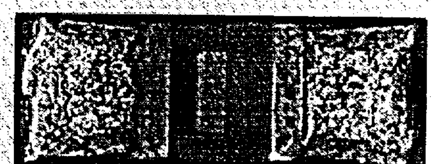


Specimen No. VT34, Test Temperature 179°F

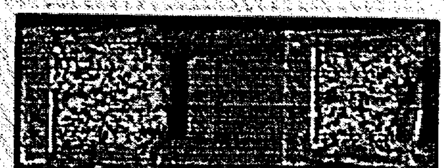
Specimen No. VT32, Test Temperature 204°F



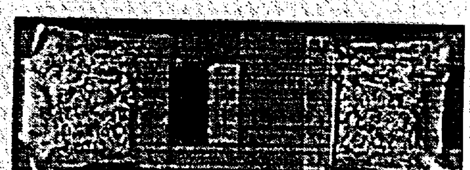
Specimen No. VT33, Test Temperature 204°F



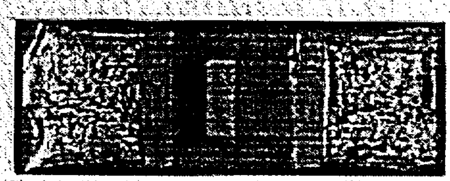
Specimen No. VT29, Test Temperature 254°F



Specimen No. VT27, Test Temperature 304°F

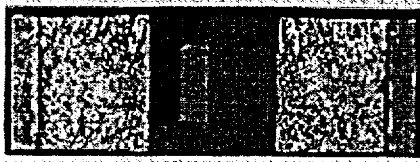


Specimen No. VT36, Test Temperature 354°F

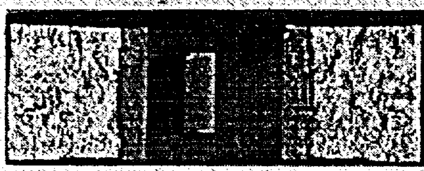


Specimen No. VT25, Test Temperature 404°F

Figure 5-14. Photographs of Charpy Impact Specimen Fracture Surfaces, Weld Metal, Weld Heat 25531 / Flux Lot 1211



Specimen No. VW36, Test Temperature -76°F



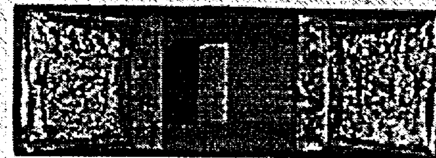
Specimen No. VW27, Test Temperature -36°F

Specimen No. VW33, Test Temperature -36°F

Specimen No. VW30, Test Temperature 4°F

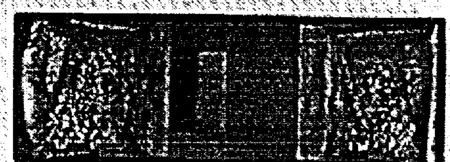
Specimen No. VW29, Test Temperature 44°F

Specimen No. VW26, Test Temperature 74°F

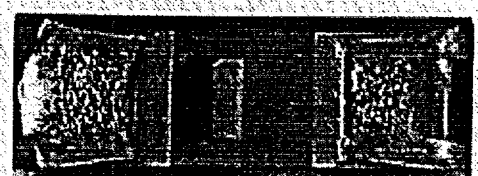


Specimen No. VW34, Test Temperature 74°F

Specimen No. VW35, Test Temperature 104°F



Specimen No. VW31, Test Temperature 129°F

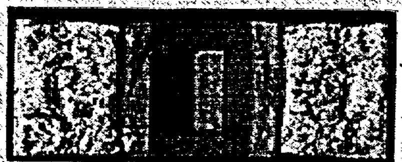


Specimen No. VW32, Test Temperature 204°F

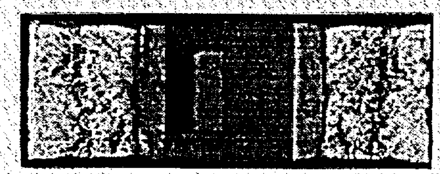
Specimen No. VW25, Test Temperature 304°F

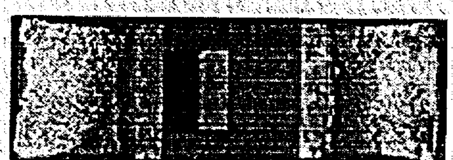
Specimen No. VW28, Test Temperature 404°F

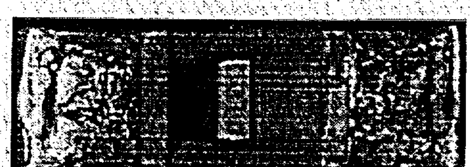
Figure 5-15. Photographs of Charpy Impact Specimen Fracture Surfaces, Heat-Affected-Zone Material


Specimen No. VH28, Test Temperature -76°F

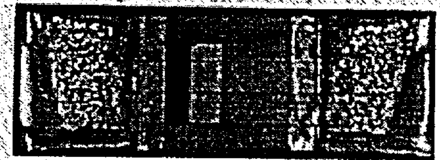

Specimen No. VH34, Test Temperature -36°F

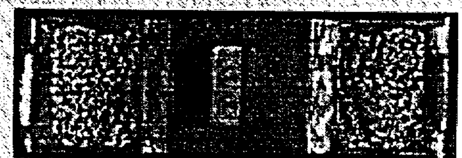

Specimen No. VH29, Test Temperature 4°F

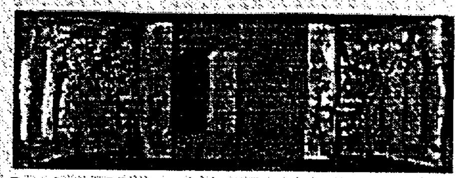

Specimen No. VH27, Test Temperature 44°F

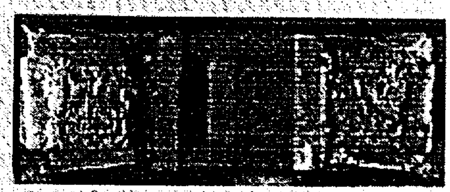

Specimen No. VH25, Test Temperature 74°F


Specimen No. VH31, Test Temperature 74°F


Specimen No. VH26, Test Temperature 104°F


Specimen No. VH36, Test Temperature 104°F


Specimen No. VH30, Test Temperature 129°F


Specimen No. VH33, Test Temperature 204°F


Specimen No. VH32, Test Temperature 304°F


Specimen No. VH35, Test Temperature 404°F

6. Neutron Fluence

6.1. Introduction

Fluence analyses as part of the reactor vessel surveillance program have three objectives:

- determine the maximum neutron fluence at the reactor vessel as a function of reactor operation
- predict the reactor vessel neutron fluence in the future, and
- determine the test specimen neutron fluence within the surveillance capsule.

Vessel fluence data is used to evaluate changes in the reference transition temperature and upper-shelf energy levels, and to establish pressure-temperature operation curves. Test specimen fluence data is used to establish a correlation between changes in material properties and fluence exposure.

Over the last fifteen years, Framatome Technologies, Inc. (FTI) has developed a calculational based fluence analysis methodology[14] that can be used to accurately predict the fast neutron 1
fluence in the reactor vessel using surveillance capsule dosimetry or cavity dosimetry (or both) to verify the fluence predictions. This methodology was developed through a full-scale benchmark experiment that was performed at the Davis-Besse Unit 1 reactor,[14] and the methodology is described in detail in Appendix F. The results of the benchmark experiment demonstrated that the accuracy of a fluence analysis that employs the FTI methodology would be unbiased and have a precision well within the NRC-suggested limit of 20%.[14], [15]

The FTI methodology was used to calculate the neutron fluence exposure to surveillance Capsule W of the North Anna Unit No. 1 nuclear reactor. The fast neutron fluences ($E > 1$ MeV) at the capsule location was calculated in accordance with the requirements of the U.S. NRC Draft Regulatory Guide DG-1053,[15] as described in detail in the FTI fluence topical report, BAW-2241P, Revision 1.[14]

The energy-dependent flux at the capsule was used to determine the calculated activity of each dosimeter. Neutron transport calculations in two-dimensional geometry were used to obtain energy dependent flux distributions throughout the core. Reactor conditions were representative of an average over the cycle 1 through 13 irradiation time period. Geometric detail is selected to explicitly represent the surveillance capsule assembly and reactor vessel. A more detailed discussion of the calculational procedure is given in Appendix F. The calculated activities were adjusted to account for known biases (photo-fission, non-saturation, and power correction), and compared directly to the measured activities. It is noted that the measurements are not used in any way to determine the magnitude of the flux or the fluence. Instead, the measurements are used only to show that the calculational results are reasonable and to show that the North Anna Unit No. 1 Capsule W results are consistent with the FTI benchmark database of uncertainties.

6.2. Capsule Fluence

The North Anna Unit No. 1 Capsule W was positioned in the reactor vessel between the thermal shield and the vessel wall with the vertical center of the capsule opposite the vertical center of the core. The capsule was located in the 25° location (as shown in Figure 6-1) for the equivalent of 5389 effective full power days (EFPDs). The rated thermal full power for cycles 1 through cycle 5 was 2775 MWt while the full power for cycle 6 was an average of 2834 MWt and cycle 7 through cycle 13 was 2893 MWt.

The E > 1.0 MeV and E > 0.1 MeV neutron fluence spectra incident on the capsule specimens were calculated for Capsule W, as shown in Table 6-1 and Table 6-2.

The E > 1.0 MeV neutron fluence spectrum is present graphically in Figure 6-2.

The E > 1.0 MeV neutron fluence at each surveillance capsule location (15°, 25°, 35°, and 45°) for each reload cycle is presented in Table 6-4. The cycle lengths for cycle 1 - 13 and the end-of-life (EOL) values are presented in Table 6-3.

The EOL values in Table 6-3 were determined assuming an EOL date of April 1, 2018 and the North Anna Unit No. 1 plant operating at a 90% capacity factor.

The results from Table 6-4 are presented graphically in Figure 6-3.

6.3. Reactor Vessel Fluence

The E > 1.0 MeV neutron fluence at the inside surface peak, ¼-thickness (¼T) peak, and ¾-thickness (¾T) peak for the 0°, 15°, 30°, and 45° locations of the North Anna Unit No. 1

reactor vessel as a function of each reload cycle is presented in Table 6-5 (inside surface peak), Table 6-6 (¼T peak), and Table 6-7 (¾T peak).

The E > 1.0 MeV extrapolated neutron fluence at the inside surface peak, ¼T peak, and ¾T peak for the 0°, 15°, 30°, and 45° locations of the North Anna Unit No. 1 reactor vessel is presented in Table 6-8.

In Table 6-8, the extrapolated fluxes are based on the cycle 11 - 13 irradiation cycles. Also, the E > 1.0 MeV fluence functions in Table 6-8 are given in the form:

$$\phi t(T) = (2.06321E+19) + \left((1.17020E+18) \times T\right)$$

where:

$\phi t(T)$	...	cumulative neutron fluence (n/cm^2) at time T (where T is in EFPY)
2.06321E+19	...	fluence at EOC 13 (n/cm^2)
1.17020E+18	...	extrapolation flux for EOC 14 – EOL (n/cm^2-year)

The results from Table 6-8 are presented graphically in Figure 6-4, Figure 6-5, and Figure 6-6.

The E > 1.0 MeV neutron fluence at the nozzle-belt forging inner surface is presented in Table 6-9. The nozzle-belt forging inner surface is located 13.6 inches above the active fuel region.

Finally, the axial and radial dependence of the pressure vessel fluence, relative to the mid-plane inside surface, is presented in Table 6-10, Table 6-11, Figure 6-7, and Figure 6-8.

6.4. Dosimetry Activity

The ratio of the calculated specific activities to the measured specific activities[10] (C/M) is presented in Table 6-12. In Table 6-12, the capsule average C/M is the average C/M for the entire North Anna Unit No. 1 Capsule W.

FRAMATOME TECHNOLOGIES

Table 6-1. Neutron Flux and Fluence Spectrum (E > 1.0 MeV) at the Center of North Anna Unit No. 1 Capsule W

Energy Group	Upper Energy (MeV)	E > 1.0 MeV Neutron Flux (n/cm^2-sec)	E > 1.0 MeV Neutron Fluence (n/cm^2)
1	1.7332E+01	1.4072E+07	6.5521E+15
2	1.4191E+01	4.0000E+07	1.8624E+16
3	1.2214E+01	1.5784E+08	7.3493E+16
4	1.0000E+01	2.9687E+08	1.3823E+17
5	8.6071E+00	4.9713E+08	2.3147E+17
6	7.4082E+00	1.1729E+09	5.4611E+17
7	6.0653E+00	1.7102E+09	7.9629E+17
8	4.9659E+00	3.1470E+09	1.4653E+18
9	3.6788E+00	2.4286E+09	1.1308E+18
10	3.0119E+00	1.8451E+09	8.5908E+17
11	2.7253E+00	2.1143E+09	9.8443E+17
12	2.4660E+00	1.0350E+09	4.8193E+17
13	2.3653E+00	2.9562E+08	1.3764E+17
14	2.3457E+00	1.4175E+09	6.5998E+17
15	2.2313E+00	3.7533E+09	1.7476E+18
16	1.9205E+00	4.1091E+09	1.9133E+18
17	1.6530E+00	5.9615E+09	2.7757E+18
18	1.3534E+00	1.0330E+10	4.8095E+18
19	1.0026E+00	8.1456E+07	3.7927E+16

Table 6-2. Neutron Flux and Fluence Spectrum (E > 0.1 MeV) at the Center of North Anna Unit No. 1 Capsule W

Energy Group	Upper Energy (MeV)	E > 0.1 MeV Neutron Flux (n/cm^2-sec)	E > 0.1 MeV Neutron Fluence (n/cm^2)
1	1.7332E+01	1.4072E+07	6.5521E+15
2	1.4191E+01	4.0000E+07	1.8624E+16
3	1.2214E+01	1.5784E+08	7.3493E+16
4	1.0000E+01	2.9687E+08	1.3823E+17
5	8.6071E+00	4.9713E+08	2.3147E+17
6	7.4082E+00	1.1729E+09	5.4611E+17
7	6.0653E+00	1.7102E+09	7.9629E+17
8	4.9659E+00	3.1470E+09	1.4653E+18
9	3.6788E+00	2.4286E+09	1.1308E+18
10	3.0119E+00	1.8451E+09	8.5908E+17
11	2.7253E+00	2.1143E+09	9.8443E+17
12	2.4660E+00	1.0350E+09	4.8193E+17
13	2.3653E+00	2.9562E+08	1.3764E+17
14	2.3457E+00	1.4175E+09	6.5998E+17
15	2.2313E+00	3.7533E+09	1.7476E+18
16	1.9205E+00	4.1091E+09	1.9133E+18
17	1.6530E+00	5.9615E+09	2.7757E+18
18	1.3534E+00	1.0330E+10	4.8095E+18
19	1.0026E+00	6.3022E+09	2.9344E+18
20	8.2085E-01	3.5746E+09	1.6644E+18
21	7.4274E-01	8.6852E+09	4.0439E+18
22	6.0810E-01	7.3186E+09	3.4076E+18
23	4.9787E-01	8.3630E+09	3.8939E+18
24	3.6883E-01	6.9421E+09	3.2323E+18
25	2.9721E-01	1.0884E+10	5.0677E+18
26	1.8316E-01	9.9358E+09	4.6262E+18
27	1.1109E-01	1.6697E+09	7.7742E+17

Table 6-3. North Anna Unit No. 1 Cycle Lengths

Cycle	EFPD	EFPY	EFPS	Cumulative EFPY
1	413	1.1307	3.5683E+07	1.1307
2	279	0.7639	2.4106E+07	1.8946
3	347	0.9500	2.9981E+07	2.8446
4	350	0.9582	3.0240E+07	3.8029
5	349	0.9555	3.0154E+07	4.7584
6	401	1.0979	3.4646E+07	5.8563
7	423	1.1581	3.6547E+07	7.0144
8	485	1.3279	4.1904E+07	8.3422
9	503	1.3771	4.3459E+07	9.7194
10	494	1.3525	4.2682E+07	11.0719
11	474	1.2977	4.0954E+07	12.3696
12	419	1.1472	3.6202E+07	13.5168
13	452	1.2375	3.9053E+07	14.7543
Cycles 14 through EOL	6383	17.4750	5.5147E+08	32.2293
		TOTAL:	4.6561E+08	

Table 6-4. E > 1.0 MeV Neutron Fluence at the 15°, 25°, 35°, and 45° Capsule Locations for North Anna Unit No. 1

Cycle	15° Capsule Location Neutron Flux (n/cm^2-sec)	15° Capsule Location Neutron Fluence (n/cm^2)	15° Capsule Location Cumulative Fluence (n/cm^2)
1	7.03258E+10	2.50945E+18	2.50945E+18
2	8.22936E+10	1.98374E+18	4.49319E+18
3	5.37861E+10	1.61255E+18	6.10574E+18
4	5.99914E+10	1.81414E+18	7.91988E+18
5	4.47085E+10	1.34812E+18	9.26800E+18
6	5.05221E+10	1.75041E+18	1.10184E+19
7	5.16233E+10	1.88669E+18	1.29051E+19
8	4.66531E+10	1.95495E+18	1.48600E+19
9	4.79910E+10	2.08565E+18	1.69457E+19
10	4.44792E+10	1.89844E+18	1.88441E+19
11	4.34429E+10	1.77914E+18	2.06233E+19
12	4.48328E+10	1.62302E+18	2.22463E+19
13	4.87556E+10	1.90404E+18	2.41503E+19

Cycle	25° Capsule Location Neutron Flux (n/cm^2-sec)	25° Capsule Location Neutron Fluence (n/cm^2)	25° Capsule Location Cumulative Fluence (n/cm^2)
1	5.82714E+10	2.07931E+18	2.07931E+18
2	6.61649E+10	1.59495E+18	3.67425E+18
3	4.52053E+10	1.35529E+18	5.02955E+18
4	4.32535E+10	1.30799E+18	6.33753E+18
5	3.72312E+10	1.12265E+18	7.46019E+18
6	4.42370E+10	1.53265E+18	8.99284E+18
7	4.37186E+10	1.59779E+18	1.05906E+19
8	4.13484E+10	1.73267E+18	1.23233E+19
9	4.05032E+10	1.76024E+18	1.40835E+19
10	3.94823E+10	1.68517E+18	1.57687E+19
11	4.01221E+10	1.64314E+18	1.74118E+19
12	3.95285E+10	1.43099E+18	1.88428E+19
13	4.28174E+10	1.67214E+18	2.05150E+19

Table 6-4. (cont'd.) E > 1.0 MeV Neutron Fluence at the 15°, 25°, 35°, and 45° Capsule Locations for North Anna Unit No. 1

Cycle	35° Capsule Location Neutron Flux (n/cm²-sec)	35° Capsule Location Neutron Fluence (n/cm²)	35° Capsule Location Cumulative Fluence (n/cm²)
1	3.38377E+10	1.20744E+18	1.20744E+18
2	3.75175E+10	9.04382E+17	2.11182E+18
3	2.47176E+10	7.41055E+17	2.85287E+18
4	2.26192E+10	6.84003E+17	3.53688E+18
5	2.19300E+10	6.61269E+17	4.19815E+18
6	2.46444E+10	8.53840E+17	5.05199E+18
7	2.58351E+10	9.44200E+17	5.99619E+18
8	2.42727E+10	1.01712E+18	7.01331E+18
9	2.38618E+10	1.03702E+18	8.05032E+18
10	2.34470E+10	1.00076E+18	9.05108E+18
11	2.46378E+10	1.00901E+18	1.00601E+19
12	2.31746E+10	8.38957E+17	1.08990E+19
13	2.48430E+10	9.70187E+17	1.18692E+19

Cycle	45° Capsule Location Neutron Flux (n/cm²-sec)	45° Capsule Location Neutron Fluence (n/cm²)	45° Capsule Location Cumulative Fluence (n/cm²)
1	2.65869E+10	9.48707E+17	9.48707E+17
2	2.84206E+10	6.85095E+17	1.63380E+18
3	1.91302E+10	5.73539E+17	2.20734E+18
4	1.75113E+10	5.29543E+17	2.73688E+18
5	1.76770E+10	5.33026E+17	3.26991E+18
6	1.89478E+10	6.56471E+17	3.92638E+18
7	2.06157E+10	7.53446E+17	4.67983E+18
8	1.90746E+10	7.99304E+17	5.47913E+18
9	1.88250E+10	8.18119E+17	6.29725E+18
10	1.87143E+10	7.98757E+17	7.09601E+18
11	1.90318E+10	7.79422E+17	7.87543E+18
12	1.80373E+10	6.52977E+17	8.52841E+18
13	1.92789E+10	7.52895E+17	9.28130E+18

Table 6-5. E > 1.0 MeV Inside Surface Peak Neutron Fluence at the North Anna Unit No. 1 Reactor Vessel 0°, 15°, 30°, and 45° Locations

Cycle	Inside Surface Peak at 0°		
	Neutron Flux (n/cm²-sec)	Neutron Fluence (n/cm²)	Cumulative Fluence (n/cm²)
1	6.43993E+10	2.29797E+18	2.29797E+18
2	7.64406E+10	1.84265E+18	4.14062E+18
3	5.01348E+10	1.50308E+18	5.64370E+18
4	5.69905E+10	1.72339E+18	7.36709E+18
5	4.02111E+10	1.21251E+18	8.57960E+18
6	4.33875E+10	1.50322E+18	1.00828E+19
7	4.09471E+10	1.49650E+18	1.15793E+19
8	3.59571E+10	1.50674E+18	1.30861E+19
9	4.05685E+10	1.76308E+18	1.48491E+19
10	3.45300E+10	1.47380E+18	1.63229E+19
11	3.75037E+10	1.53591E+18	1.78588E+19
12	3.49990E+10	1.26702E+18	1.91259E+19
13	3.85688E+10	1.50622E+18	2.06321E+19

Cycle	Inside Surface Peak at 15°		
	Neutron Flux (n/cm²-sec)	Neutron Fluence (n/cm²)	Cumulative Fluence (n/cm²)
1	3.23987E+10	1.15609E+18	1.15609E+18
2	3.78998E+10	9.13597E+17	2.06969E+18
3	2.48153E+10	7.43982E+17	2.81367E+18
4	2.75080E+10	8.31840E+17	3.64551E+18
5	2.05586E+10	6.19915E+17	4.26542E+18
6	2.32186E+10	8.04439E+17	5.06986E+18
7	2.35926E+10	8.62244E+17	5.93211E+18
8	2.13211E+10	8.93439E+17	6.82555E+18
9	2.20270E+10	9.57275E+17	7.78282E+18
10	2.03589E+10	8.68949E+17	8.65177E+18
11	2.00113E+10	8.19535E+17	9.47130E+18
12	2.05192E+10	7.42827E+17	1.02141E+19
13	2.23094E+10	8.71245E+17	1.10854E+19

Table 6-5. (cont'd.) E > 1.0 MeV Inside Surface Peak Neutron Fluence at the North Anna Unit No. 1 Reactor Vessel 0°, 15°, 30°, and 45° Locations

	Inside Surface Peak at 30°		
Cycle	Neutron Flux (n/cm²-sec)	Neutron Fluence (n/cm²)	Cumulative Fluence (n/cm²)
1	1.87158E+10	6.67839E+17	6.67839E+17
2	2.09816E+10	5.05773E+17	1.17361E+18
3	1.41552E+10	4.24383E+17	1.59800E+18
4	1.31507E+10	3.97679E+17	1.99567E+18
5	1.20502E+10	3.63356E+17	2.35903E+18
6	1.40766E+10	4.87704E+17	2.84673E+18
7	1.42023E+10	5.19053E+17	3.36579E+18
8	1.34233E+10	5.62490E+17	3.92828E+18
9	1.31385E+10	5.70991E+17	4.49927E+18
10	1.28647E+10	5.49084E+17	5.04835E+18
11	1.34050E+10	5.48983E+17	5.59733E+18
12	1.27735E+10	4.62422E+17	6.05976E+18
13	1.38124E+10	5.39411E+17	6.59917E+18

	Inside Surface Peak at 45°		
Cycle	Neutron Flux (n/cm²-sec)	Neutron Fluence (n/cm²)	Cumulative Fluence (n/cm²)
1	1.25287E+10	4.47063E+17	4.47063E+17
2	1.34089E+10	3.23230E+17	7.70292E+17
3	9.10381E+09	2.72940E+17	1.04323E+18
4	8.31075E+09	2.51317E+17	1.29455E+18
5	8.38202E+09	2.52748E+17	1.54730E+18
6	9.01818E+09	3.12447E+17	1.85974E+18
7	1.42023E+10	5.19053E+17	2.37880E+18
8	9.05762E+09	3.79551E+17	2.75835E+18
9	8.94114E+09	3.88575E+17	3.14692E+18
10	8.88589E+09	3.79264E+17	3.52619E+18
11	9.05525E+09	3.70845E+17	3.89703E+18
12	8.58663E+09	3.10850E+17	4.20788E+18
13	9.16730E+09	3.58009E+17	4.56589E+18

Table 6-6. E > 1.0 MeV ¼T Location Peak Neutron Fluence at the North Anna Unit No. 1 Reactor Vessel 0°, 15°, 30°, and 45° Locations

	¼T Peak at 0°		
Cycle	Neutron Flux (n/cm^2-sec)	Neutron Fluence (n/cm^2)	Cumulative Fluence (n/cm^2)
1	4.03526E+10	1.43991E+18	1.43991E+18
2	4.79029E+10	1.15473E+18	2.59464E+18
3	3.13314E+10	9.39341E+17	3.53398E+18
4	3.56512E+10	1.07809E+18	4.61207E+18
5	2.51610E+10	7.58696E+17	5.37077E+18
6	2.71986E+10	9.42332E+17	6.31310E+18
7	2.57405E+10	9.40745E+17	7.25385E+18
8	2.25928E+10	9.46729E+17	8.20057E+18
9	2.54299E+10	1.10516E+18	9.30574E+18
10	2.16853E+10	9.25561E+17	1.02313E+19
11	2.35003E+10	9.62420E+17	1.11937E+19
12	2.19654E+10	7.95181E+17	1.19889E+19
13	2.42127E+10	9.45575E+17	1.29345E+19

	¼T Peak at 15°		
Cycle	Neutron Flux (n/cm^2-sec)	Neutron Fluence (n/cm^2)	Cumulative Fluence (n/cm^2)
1	2.06484E+10	7.36800E+17	7.36800E+17
2	2.41394E+10	5.81895E+17	1.31870E+18
3	1.57997E+10	4.73689E+17	1.79238E+18
4	1.75213E+10	5.29843E+17	2.32223E+18
5	1.30976E+10	3.94938E+17	2.71717E+18
6	1.47768E+10	5.11962E+17	3.22913E+18
7	1.50038E+10	5.48347E+17	3.77747E+18
8	1.35566E+10	5.68077E+17	4.34555E+18
9	1.40137E+10	6.09022E+17	4.95457E+18
10	1.29457E+10	5.52542E+17	5.50712E+18
11	1.27510E+10	5.22197E+17	6.02931E+18
12	1.30381E+10	4.72002E+17	6.50131E+18
13	1.41837E+10	5.53914E+17	7.05523E+18

Table 6-6. (cont'd). E > 1.0 MeV ¼T Location Peak Neutron Fluence at the North Anna Unit No. 1 Reactor Vessel 0°, 15°, 30°, and 45° Locations

	¼T Peak at 30°		
Cycle	Neutron Flux (n/cm^2-sec)	Neutron Fluence (n/cm^2)	Cumulative Fluence (n/cm^2)
1	1.17711E+10	4.20030E+17	4.20030E+17
2	1.31795E+10	3.17701E+17	7.37731E+17
3	8.90804E+09	2.67070E+17	1.00480E+18
4	8.28823E+09	2.50636E+17	1.25544E+18
5	7.59227E+09	2.28934E+17	1.48437E+18
6	8.85064E+09	3.06643E+17	1.79101E+18
7	8.93840E+09	3.26674E+17	2.11769E+18
8	8.44700E+09	3.53963E+17	2.47165E+18
9	8.27731E+09	3.59725E+17	2.83138E+18
10	8.10322E+09	3.45858E+17	3.17723E+18
11	8.42483E+09	3.45027E+17	3.52226E+18
12	8.04311E+09	2.91173E+17	3.81343E+18
13	8.69764E+09	3.39667E+17	4.15310E+18

	¼T Peak at 45°		
Cycle	Neutron Flux (n/cm^2-sec)	Neutron Fluence (n/cm^2)	Cumulative Fluence (n/cm^2)
1	7.95678E+09	2.83923E+17	2.83923E+17
2	8.52105E+09	2.05405E+17	4.89328E+17
3	5.78857E+09	1.73546E+17	6.62874E+17
4	5.28548E+09	1.59833E+17	8.22707E+17
5	5.32630E+09	1.60607E+17	9.83314E+17
6	5.73527E+09	1.98706E+17	1.18202E+18
7	8.93840E+09	3.26674E+17	1.50869E+18
8	5.76433E+09	2.41548E+17	1.75024E+18
9	5.69021E+09	2.47292E+17	1.99754E+18
10	5.65164E+09	2.41221E+17	2.23876E+18
11	5.76461E+09	2.36082E+17	2.47484E+18
12	5.46532E+09	1.97853E+17	2.67269E+18
13	5.83350E+09	2.27815E+17	2.90051E+18

Table 6-7. E > 1.0 MeV ¾T Location Peak Neutron Fluence at the North Anna Unit No. 1 Reactor Vessel 0°, 15°, 30°, and 45° Locations

	¾T Peak at 0°		
Cycle	Neutron Flux (n/cm²-sec)	Neutron Fluence (n/cm²)	Cumulative Fluence (n/cm²)
1	8.57639E+09	3.06033E+17	3.06033E+17
2	1.01793E+10	2.45378E+17	5.51411E+17
3	6.62565E+09	1.98642E+17	7.50054E+17
4	7.55290E+09	2.28400E+17	9.78453E+17
5	5.33796E+09	1.60959E+17	1.13941E+18
6	5.78618E+09	2.00470E+17	1.33988E+18
7	5.50875E+09	2.01330E+17	1.54121E+18
8	4.83438E+09	2.02580E+17	1.74379E+18
9	5.41617E+09	2.35382E+17	1.97917E+18
10	4.63850E+09	1.97979E+17	2.17715E+18
11	4.99996E+09	2.04766E+17	2.38192E+18
12	4.69274E+09	1.69885E+17	2.55180E+18
13	5.17206E+09	2.01984E+17	2.75379E+18

	¾T Peak at 15°		
Cycle	Neutron Flux (n/cm²-sec)	Neutron Fluence (n/cm²)	Cumulative Fluence (n/cm²)
1	4.61386E+09	1.64637E+17	1.64637E+17
2	5.39070E+09	1.29946E+17	2.94583E+17
3	3.53341E+09	1.05934E+17	4.00518E+17
4	3.91308E+09	1.18332E+17	5.18849E+17
5	2.92600E+09	8.82295E+16	6.07079E+17
6	3.29781E+09	1.14257E+17	7.21336E+17
7	3.33692E+09	1.21955E+17	8.43291E+17
8	3.01480E+09	1.26332E+17	9.69624E+17
9	3.12270E+09	1.35710E+17	1.10533E+18
10	2.87867E+09	1.22866E+17	1.22820E+18
11	2.85033E+09	1.16731E+17	1.34493E+18
12	2.89813E+09	1.04917E+17	1.44985E+18
13	3.15435E+09	1.23186E+17	1.57303E+18

Table 6-7. (cont'd). E > 1.0 MeV ¾T Location Peak Neutron Fluence at the North Anna Unit No. 1 Reactor Vessel 0°, 15°, 30°, and 45° Locations

	¾T Peak at 30°		
Cycle	Neutron Flux (n/cm^2-sec)	Neutron Fluence (n/cm^2)	Cumulative Fluence (n/cm^2)
1	2.58109E+09	9.21015E+16	9.21015E+16
2	2.88710E+09	6.95953E+16	1.61697E+17
3	1.95594E+09	5.86407E+16	2.20338E+17
4	1.83235E+09	5.54101E+16	2.75748E+17
5	1.66919E+09	5.03320E+16	3.26080E+17
6	1.93733E+09	6.71217E+16	3.93201E+17
7	1.95868E+09	7.15843E+16	4.64786E+17
8	1.84989E+09	7.75178E+16	5.42303E+17
9	1.81320E+09	7.88002E+16	6.21104E+17
10	1.77358E+09	7.56991E+16	6.96803E+17
11	1.83762E+09	7.52573E+16	7.72060E+17
12	1.76457E+09	6.38803E+16	8.35940E+17
13	1.90723E+09	7.44827E+16	9.10423E+17

	¾T Peak at 45°		
Cycle	Neutron Flux (n/cm^2-sec)	Neutron Fluence (n/cm^2)	Cumulative Fluence (n/cm^2)
1	1.78061E+09	6.35380E+16	6.35380E+16
2	1.91229E+09	4.60969E+16	1.09635E+17
3	1.30305E+09	3.90665E+16	1.48701E+17
4	1.19175E+09	3.60385E+16	1.84740E+17
5	1.19488E+09	3.60298E+16	2.20770E+17
6	1.29094E+09	4.47265E+16	2.65496E+17
7	1.95868E+09	7.15843E+16	3.37080E+17
8	1.29536E+09	5.42810E+16	3.91361E+17
9	1.27875E+09	5.55734E+16	4.46935E+17
10	1.26861E+09	5.41462E+16	5.01081E+17
11	1.29595E+09	5.30739E+16	5.54155E+17
12	1.22982E+09	4.45213E+16	5.98676E+17
13	1.31292E+09	5.12733E+16	6.49949E+17

Table 6-8. Extrapolated E > 1.0 MeV Neutron Fluence at the North Anna Unit No. 1 Reactor Vessel 0°, 15°, 30°, and 45° Locations

Inside Surface Peak				
Location	Extrapolated Flux (n/cm^2-year)	Fluence at EOC 13 (n/cm^2)	Cycle 14 through EOL Fluence Function	EOL Fluence (n/cm^2)
0°	1.17020E+18	2.06321E+19	$\phi t(T)$ = 2.06321E19 + (1.17020E18 × T)	4.10813E+19
15°	6.60873E+17	1.10854E+19	$\phi t(T)$ = 1.10854E19 + (6.60873E17 × T)	2.26342E+19
30°	4.21142E+17	6.59917E+18	$\phi t(T)$ = 6.59917E18 + (4.21142E17 × T)	1.39586E+19
45°	2.82343E+17	4.56589E+18	$\phi t(T)$ = 4.56589E18 + (2.82343E17 × T)	9.49985E+18

¼T Peak				
Location	Extrapolated Flux (n/cm^2-year)	Fluence at EOC 13 (n/cm^2)	Cycle 14 through EOL Fluence Function	EOL Fluence (n/cm^2)
0°	7.34078E+17	1.29345E+19	$\phi t(T)$ = 1.29345E19 + (7.34078E17 × T)	2.57625E+19
15°	4.20408E+17	7.05523E+18	$\phi t(T)$ = 7.05523E18 + (4.20408E17 × T)	1.44019E+19
30°	2.65008E+17	4.15310E+18	$\phi t(T)$ = 4.15310E18 + (2.65008E17 × T)	8.78412E+18
45°	1.79706E+17	2.90051E+18	$\phi t(T)$ = 2.90051E18 + (1.79706E17 × T)	6.04086E+18

¾T Peak				
Location	Extrapolated Flux (n/cm^2-year)	Fluence at EOC 13 (n/cm^2)	Cycle 14 through EOL Fluence Function	EOL Fluence (n/cm^2)
0°	1.56361E+17	2.75379E+18	$\phi t(T)$ = 2.75379E18 + (1.56361E17 × T)	5.48620E+18
15°	9.35057E+16	1.57303E+18	$\phi t(T)$ = 1.57303E18 + (9.35057E16 × T)	3.20705E+18
30°	5.79256E+16	9.10423E+17	$\phi t(T)$ = 9.10423E17 + (5.79256E16 × T)	1.92267E+18
45°	4.03673E+16	6.49949E+17	$\phi t(T)$ = 6.49949E17 + (4.03673E16 × T)	1.35537E+18

Table 6-9. E > 1.0 MeV Peak Neutron Fluence at the North Anna Unit No. 1 Reactor Vessel Nozzle-Belt Forging

Cycle	Peak Fluence on Nozzle-Belt Inner Surface Neutron Flux (n/cm^2-sec)	Neutron Fluence (n/cm^2)	Cumulative Fluence (n/cm^2)
1	6.08943E+09	2.17290E+17	2.17290E+17
2	7.22802E+09	1.74236E+17	3.91526E+17
3	4.74062E+09	1.42127E+17	5.33653E+17
4	5.38887E+09	1.62959E+17	6.96613E+17
5	3.80225E+09	1.14652E+17	8.11265E+17
6	4.10261E+09	1.42141E+17	9.53405E+17
7	3.87185E+09	1.41505E+17	1.09491E+18
8	3.40001E+09	1.42474E+17	1.23738E+18
9	3.83605E+09	1.66712E+17	1.40410E+18
10	3.26507E+09	1.39358E+17	1.54345E+18
11	3.54625E+09	1.45232E+17	1.68869E+18
12	3.30941E+09	1.19806E+17	1.80849E+18
13	3.64697E+09	1.42424E+17	1.95092E+18

Nozzle-Belt Inner Surface Peak				
Location	Extrapolated Flux (n/cm^2-year)	Fluence at EOC 13 (n/cm^2)	Cycle 14 through EOL Fluence Function	EOL Fluence (n/cm^2)
0°	1.10651E+17	1.95092E+18	$\phi t(T) = 1.95092E18 + (1.10651E17 \times T)$	3.88454E+18

Table 6-10. Radial Dependence of the Pressure Vessel Neutron Fluence (Relative to the Mid-Plane Inside Surface)

Distance from Core Centerline (cm)	Pressure Vessel Radial Fluence Relative to Mid-Plane IS
200.090	1.00000
200.890	0.93860
202.390	0.80627
204.390	0.63465
206.390	0.48865
208.390	0.37227
209.890	0.30029
211.340	0.24597
213.290	0.18603
215.290	0.13859
217.340	0.10166
219.315	0.07366

Table 6-11. Axial Dependence of the Pressure Vessel Neutron Fluence (Relative to the Mid-Plane Inside Surface)

Distance from Bottom of Active Fuel (cm)	Pressure Vessel Axial Fluence Relative to Mid-Plane IS	Distance from Bottom of Active Fuel (cm)	Pressure Vessel Axial Fluence Relative to Mid-Plane IS
0.000	0.33180	57.299	1.03239
1.152	0.34830	62.069	1.03424
2.303	0.36805	66.839	1.03275
3.455	0.38927	71.609	1.03194
4.607	0.41235	76.379	1.01396
5.758	0.43545	81.149	0.98073
6.905	0.45653	85.919	0.96275
8.048	0.47636	89.815	0.95937
9.190	0.50109	92.718	0.95844
10.333	0.51475	95.502	0.95748
11.809	0.53879	99.475	0.95516
13.970	0.58098	104.636	0.96214
16.483	0.62477	109.797	0.98514
18.995	0.66866	114.958	1.01649
21.326	0.71263	120.119	1.01962
23.838	0.75929	125.279	1.01717
26.532	0.80487	130.440	1.02095
29.044	0.83837	135.601	1.01314
31.556	0.86877	140.762	0.98453
34.069	0.89587	145.923	0.96909
36.581	0.92359	151.084	0.96091
39.093	0.94434	155.579	0.95899
41.606	0.96267	158.873	0.96328
44.118	0.98162	162.018	0.95745
47.759	0.99957	165.550	0.96165
52.529	1.01759	169.772	0.97869

Table 6-11. (cont'd.) Axial Dependence of the Pressure Vessel Neutron Fluence (Relative to the Mid-Plane Inside Surface)

Distance from Bottom of Active Fuel (cm)	Pressure Vessel Axial Fluence Relative to Mid-Plane IS	Distance from Bottom of Active Fuel (cm)	Pressure Vessel Axial Fluence Relative to Mid-Plane IS
174.683	0.99715	306.863	0.94470
179.594	1.00000	310.940	0.93057
184.505	1.00130	315.018	0.92011
189.416	0.99759	319.096	0.91710
194.327	0.97585	322.319	0.90955
199.238	0.96295	324.688	0.90026
204.149	0.96111	327.057	0.88026
208.514	0.96206	329.426	0.84951
212.905	0.96202	331.794	0.82591
217.866	0.96814	334.163	0.79541
222.827	0.97983	336.532	0.76657
227.788	0.99955	338.901	0.74242
232.749	1.01260	341.270	0.71567
237.711	1.00367	344.364	0.67472
242.672	0.98624	347.441	0.63196
247.633	0.97895	349.773	0.59576
252.594	0.97679	352.105	0.56102
256.984	0.97645	354.438	0.52627
261.525	0.98170	356.242	0.50359
266.788	0.98263	357.518	0.49660
272.050	0.99519	358.793	0.48574
277.313	1.01895	360.069	0.47475
282.575	1.01719	361.344	0.46010
287.838	0.99083	362.620	0.43863
293.100	0.97463	363.895	0.42092
298.363	0.96564	364.696	0.41056
302.909	0.95473	365.021	0.40686

Table 6-12. North Anna Unit No. 1 Capsule W C/M Ratios

Dosimeter Identification	Measured Activity (μCi/g)	Calculated Activity (μCi/g)	C/M	Capsule Average C/M
Top Mid Cu	6.675E+00	6.740E+00	1.010	1.03
Middle Cu	7.177E+00	7.294E+00	1.016	
Bottom Mid Cu	6.908E+00	6.683E+00	0.967	
Top Mid Ni	9.231E+02	1.015E+03	1.099	
Middle Ni	1.000E+03	1.091E+03	1.091	
Bottom Mid Ni	9.426E+02	9.939E+02	1.054	
Top Fe	8.114E+02	8.761E+02	1.080	
Top Mid Fe	7.501E+02	8.129E+02	1.084	
Middle Fe	8.001E+02	8.759E+02	1.095	
Bottom Mid Fe	7.748E+02	7.984E+02	1.030	
Bottom Fe	7.720E+02	8.249E+02	1.069	
Sh U-238	2.014E+01	1.929E+01	0.958	

Figure 6-1. North Anna Unit No. 1 Capsule W Location Plan View

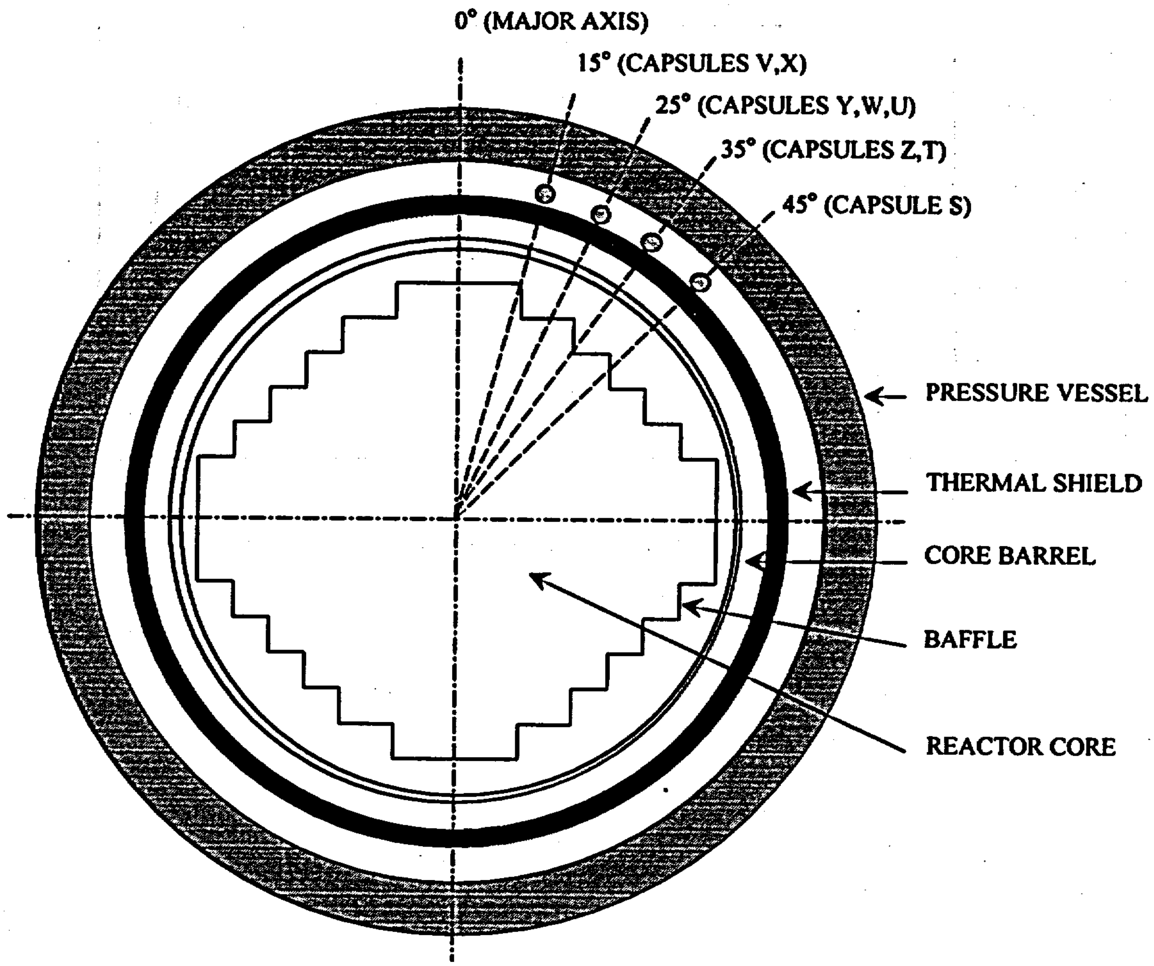


Figure 6-2. Relative E > 1.0 MeV Neutron Fluence

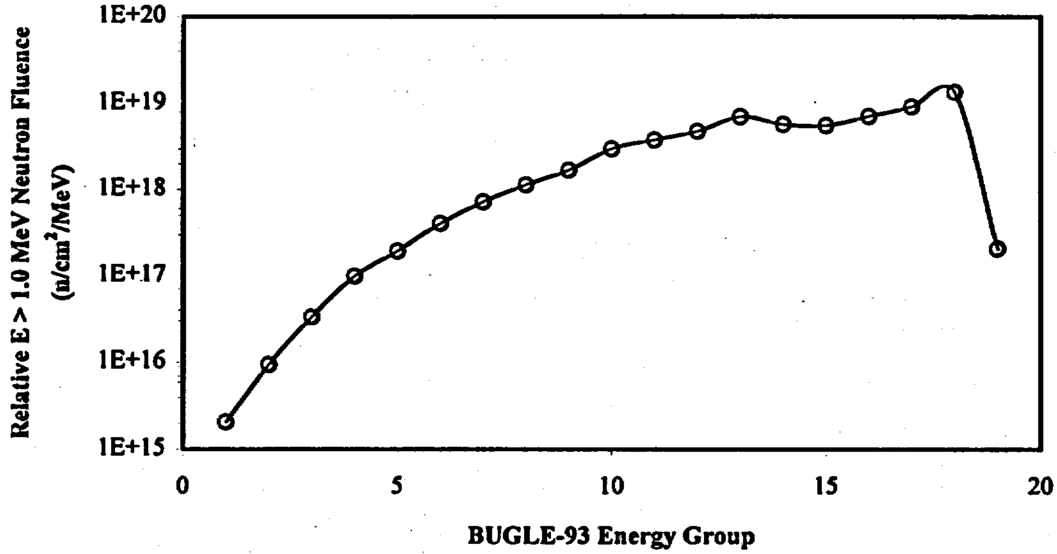
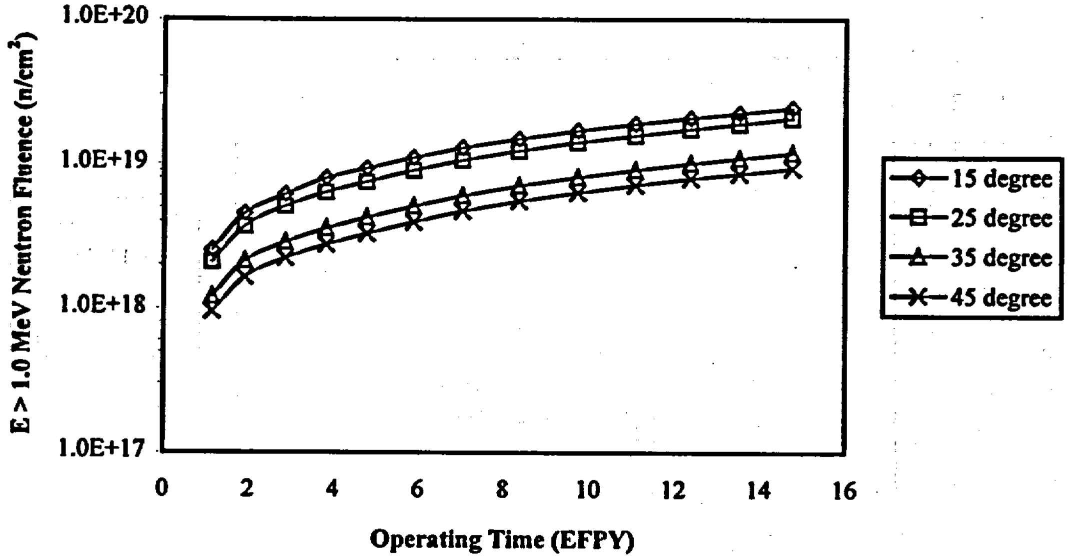
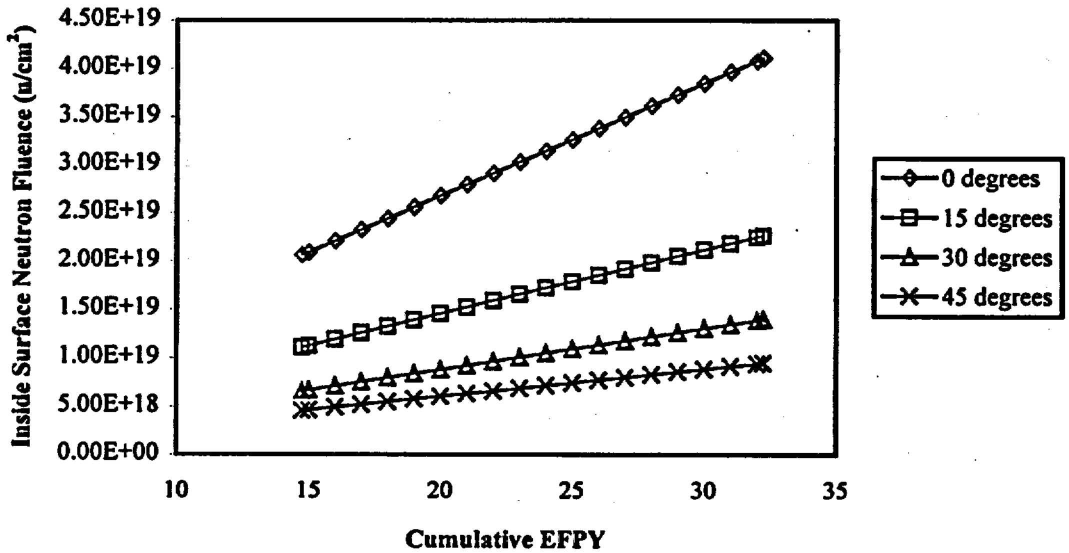


Figure 6-3. E > 1.0 MeV Neutron Fluence at the 15°, 25°, 35°, and 45° Capsule Locations for North Anna Unit No. 1



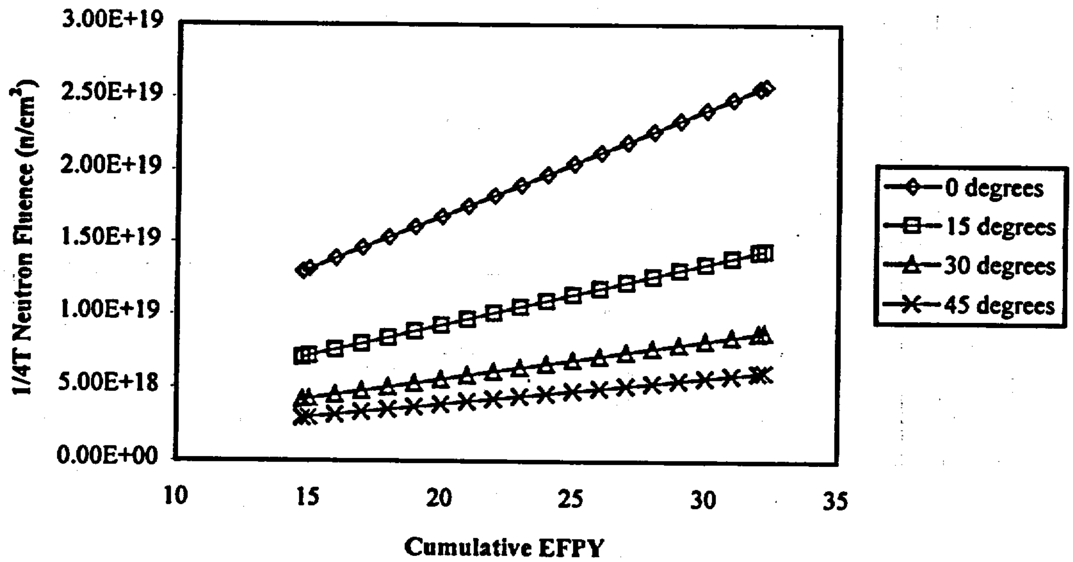
FRAMATOME TECHNOLOGIES

Figure 6-4. Extrapolated E > 1.0 MeV Inside Surface Neutron Fluence at the North Anna Unit No. 1 Reactor Vessel 0°, 15°, 30°, and 45° Locations



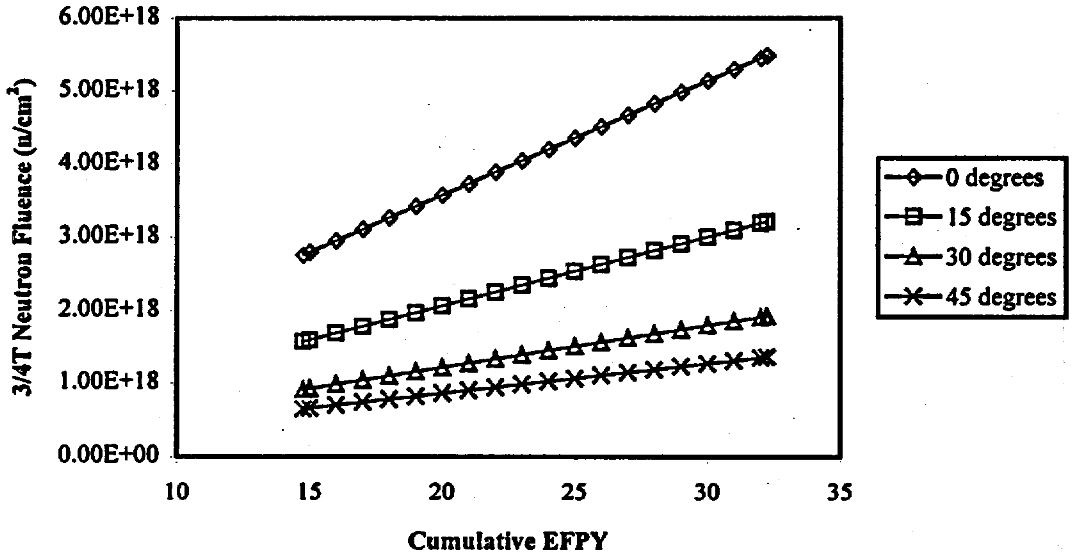
FRAMATOME TECHNOLOGIES

Figure 6-5. Extrapolated E > 1.0 MeV ¼T Neutron Fluence at the North Anna Unit No. 1 Reactor Vessel 0°, 15°, 30°, and 45° Locations



FRAMATOME TECHNOLOGIES

Figure 6-6. Extrapolated E > 1.0 MeV ¾T Neutron Fluence at the North Anna Unit No. 1 Reactor Vessel 0°, 15°, 30°, and 45° Locations



FRAMATOME TECHNOLOGIES

Figure 6-7. Radial Dependence of the Pressure Vessel Neutron Fluence (Relative to the Mid-Plane Inside Surface)

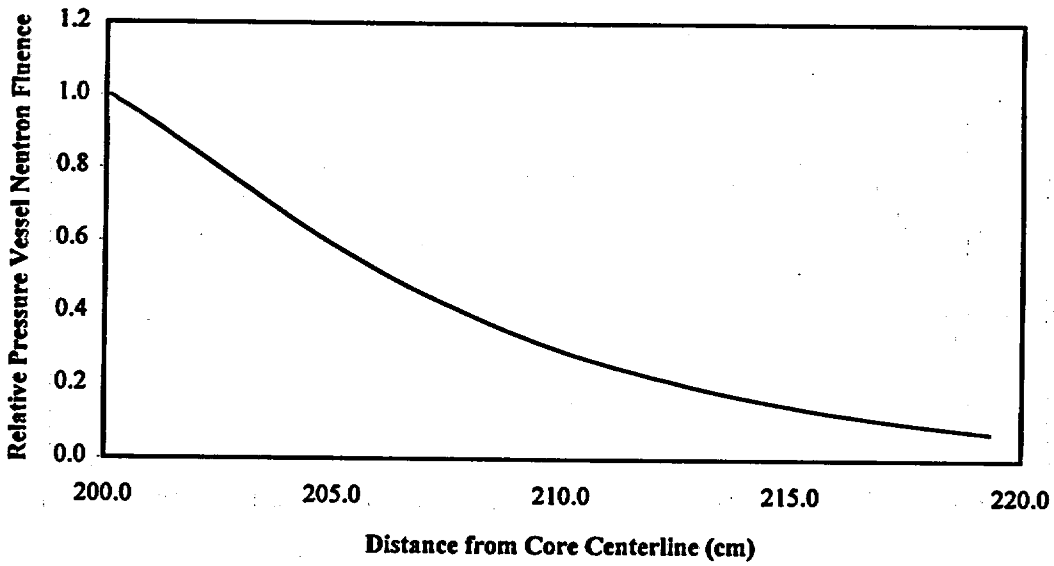
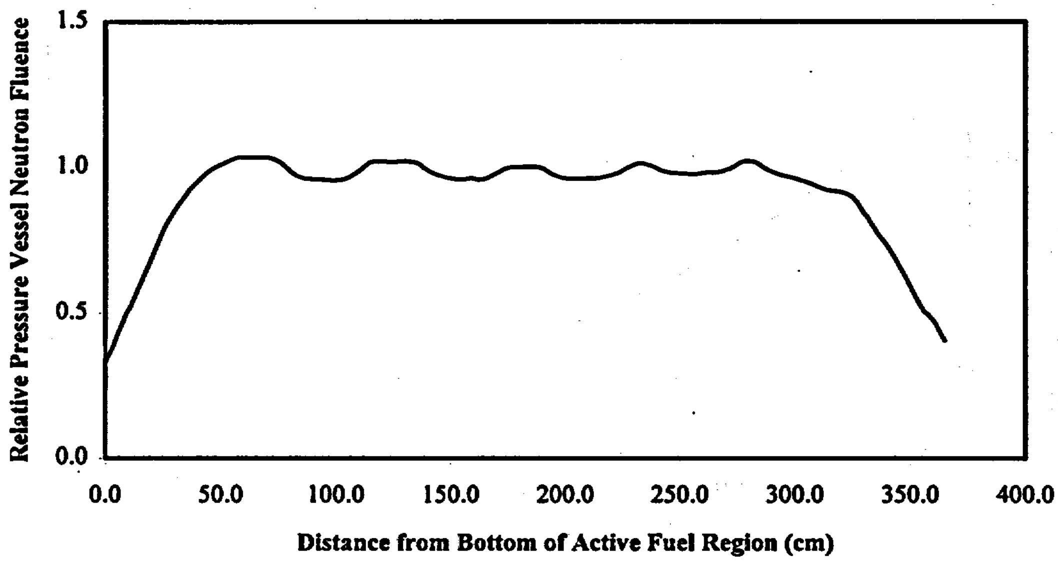


Figure 6-8. Axial Dependence of the Pressure Vessel Neutron Fluence (Relative to the Mid-Plane Inside Surface)



7. Discussion of Capsule Results

7.1. Copper and Nickel Chemical Composition Data

To date, several chemical analyses have been performed on the North Anna Unit No. 1 RVSP base metal forging and weld metal. These analyses have been performed on the unirradiated surveillance materials, broken Charpy specimens tested as part of the North Anna Unit No. 1, Capsule U analysis, and broken Charpy specimens tested as part of the North Anna Unit No. 1, Capsule W analysis. The mean copper and nickel contents for the North Anna No. 1 RVSP base metal forging and weld metal represent the best-estimate chemical contents for these materials. The copper and nickel chemical content data and their calculated means are presented in Tables 7-1 and 7-2.

7.2. Unirradiated Material Property Data

The base metal and weld metal were selected for inclusion in the North Anna Unit No. 1 surveillance program in accordance with the criteria in effect at the time the program was designed. The applicable selection criterion was based on the unirradiated properties of the North Anna Unit No. 1 reactor vessel beltline region materials only.

The unirradiated mechanical properties for the North Anna Unit No. 1 RVSP materials are summarized in Appendices C and D of this report.

7.3. Irradiated Property Data

In addition to the Capsule W mechanical test data, surveillance data is available from the North Anna Unit No. 1 RVSP Capsules V and U. Framatome Technologies, Inc. (formally Babcock & Wilcox Nuclear Power Generation Division) performed the testing and evaluation for Capsule V,[2] while the testing and evaluation for Capsule U was performed by Westinghouse Electric Corporation.[3]

The capsule fluences for the North Anna Unit No. 1 Capsule V and Capsule U have been reanalyzed, and the results have been documented in WCAP-14044.[16] Based on this re-evaluation, the capsule fluences (E > 1.0 MeV) for Capsule V and Capsule U are 2.63×10^{18} n/cm^2 and 8.72×10^{18} n/cm^2 respectively.

7.3.1. Tensile Properties

Table 7-3 compares the irradiated and unirradiated tensile properties. Review of the surveillance tensile test data indicates that the ultimate strength and yield strength changes in the base metal forging as a result of irradiation and the corresponding changes in ductility are within the ranges observed for similar irradiated materials. The changes in tensile properties for the surveillance weld metal, as a result of irradiation, are also within the observed ranges for similar irradiated materials. The general behavior of the tensile properties as a function of neutron irradiation is an increase in both ultimate and yield strength and a decrease in ductility as measured by both total elongation and reduction in area.

7.3.2. Impact Properties

Tables 7-4 and 7-5 compare the measured changes in irradiated Charpy V-notch impact properties from Capsule W with the predicted changes in accordance with Regulatory Guide 1.99, Revision 2.[17]

The measured 30 ft-lb transition temperature shifts for the surveillance base metal forging are less than the shifts predicted using Regulatory Guide 1.99, Revision 2, Position 1.1, and with the addition of the margin (σ) the predicted shifts for these materials have a large amount of conservatism. The measured 30 ft-lb transition temperature shift for the surveillance weld metal is greater than the shift predicted using Regulatory Guide 1.99, Revision 2, Position 1.1. However, this measured 30 ft-lb transition temperature shift falls within one standard deviation (σ) of the predicted shift (see Table 7-4).

The measured upper-shelf energies for the North Anna Unit No. 1 Capsule W surveillance materials do not fall below the required 50 ft-lb limit. The measured percent decrease in C_vUSE for the measured surveillance base metal forging and the surveillance weld metal are in agreement with the values predicted using Regulatory Guide 1.99, Revision 2. The base metal forging in the axial orientation and the weld metal have a predicted percent decrease in C_VUSE slightly greater than the measured values. The percent reduction in C_vUSE for the base metal forging in the tangential orientation showed the best comparison between its measured data and the value predicted using Regulatory Guide 1.99, Revision 2.

The original unirradiated Charpy impact data and irradiated Charpy impact data for Capsules V and U were evaluated based on hand-fit Charpy curves generated using engineering judgement. These data were re-plotted and re-evaluated herein using a hyperbolic tangent curve-fitting program to be consistent with the Capsule W Charpy curves and evaluation. The results of the re-evaluation are presented in Appendix D. In addition, Appendix E contains a comparison of the Charpy V-notch shift results for each surveillance material, hand-fit versus hyperbolic tangent curve-fit.

The radiation-induced changes in toughness of the North Anna Unit No. 1 surveillance materials are summarized in Table 7-6.

In addition, the Sequoyah Unit No. 1 plant specific RVSP and the Sequoyah Unit No. 2 plant specific RVSP provide data for the weld metal W05A (wire heat 25295 / flux lot 1170) and weld metal W05B (wire heat 4278 / flux lot 1211) respectively. The original unirradiated Charpy impact data and irradiated Charpy impact data for both RVSPs were evaluated based on hand-fit Charpy curves generated using engineering judgement. These data were re-plotted and re-evaluated using a hyperbolic tangent curve-fitting program to be consistent with the evaluation of the NA1 plant specific RVSP data.[18] The radiation-induced changes in toughness of the Sequoyah Unit No. 1 and Sequoyah Unit No. 2 surveillance materials are summarized in Table 7-7.

7.4. Reactor Vessel Fracture Toughness

7.4.1. Adjusted Reference Temperature Evaluation

The adjusted reference temperatures for the North Anna Unit No. 1 reactor vessel beltline region materials were calculated in accordance with Regulatory Guide 1.99, Revision 2 applicable to 32.2 effective full power years (EFPY), and the results are presented in Table 7-8. The evaluations were performed at the ¼-thickness (¼T) and ¾-thickness (¾T) wall location of each beltline material. Based on these results, the controlling beltline material for the North Anna Unit No. 1 reactor vessel is the lower shell forging (Forging 03), heat no. 990400/292332.

7.4.2. Pressurized Thermal Shock Evaluation

A pressurized thermal shock (PTS) evaluation for the North Anna Unit No. 1 reactor vessel beltline materials was performed in accordance with 10 CFR 50.61,[19] and the results are shown in Table 7-9. The results of the PTS evaluation demonstrate that the North Anna Unit No. 1 reactor vessel beltline materials will not exceed the PTS screening criteria before end-of-life (32.2 EFPY). The controlling beltline material for the North Anna Unit No. 1 reactor vessel with respect to PTS is the lower shell forging (Forging 03), heat no. 990400/292332 with a RT_{PTS} value of 184.9°F which is well below the PTS screening criterion of 270°F.

Table 7-1. Copper and Nickel Chemical Composition Data for North Anna Unit No. 1 Reactor Vessel Surveillance Base Metal Forging 03 (Heat No. 990400/292332)

Analysis Source	Cu Wt%	Ni Wt%	Reference
RVSP Baseline Chemistry (Westinghouse Analysis)	0.16	0.79	WCAP-8771 (RVSP Description)
RVSP Baseline Chemistry (Rotterdam Dockyard Analysis)	0.15	0.80	WCAP-8771 (RVSP Description)
CVN Specimen: VT-71	0.158	0.893	WCAP-11777 (Capsule U)
CVN Specimen: VT-36	0.155	0.785	Capsule W
Mean	0.156	0.817	

Table 7-2. Copper and Nickel Chemical Composition Data for North Anna Unit No. 1 Reactor Vessel Surveillance Weld Metal (Wire Heat 25531 / Flux Lot 1211)

Analysis Source	Cu Wt%	Ni Wt%	Reference
RVSP Baseline Chemistry (Westinghouse Analysis)	0.086	0.11	WCAP-8771 (RVSP Description)
CVN Specimen: VW-71	0.124	0.152	WCAP-11777 (Capsule U)
CVN Specimen: VW-29	0.084	0.11	Capsule W
Mean	0.098	0.124	

Table 7-3. Summary of North Anna Unit No. 1 Reactor Vessel Surveillance Capsules Tensile Test Results

Material	Fluence, 10^{19} n/cm^2	Test Temp., F	Strength, ksi				Ductility, %			
			Ultimate	%[a]	Yield	%[a]	Total Elong.	%[a]	Reduction of Area	%[a]
Base Metal Forging 03, Ht. No. 990400/292332 (Axial)	0.00	Room	92.6[b]	—	70.7[b]	—	18.8[b]	—	58.8[b]	—
		300	84.9[b]	—	64.4[b]	—	22.6[b]	—	62.5[b]	—
		550	86.2[b]	—	55.6[b]	—	23.3[b]	—	54.5[b]	—
	0.263	76	99.3	+7.2	80.4	+13.7	20.9	+11.2	54.9	-6.6
		548	95.7	+11.0	64.7	+16.4	18.4	-21.0	47.0	-13.8
	0.872	275	96.8	+14.0	75.9	+17.9	16.4	-27.4	46	-26.4
		550	95.7	+11.0	69.4	+24.8	13.5	-42.1	41	-24.8
	2.052	300	98.7	+16.3	78.5	+21.9	14.4	-36.3	47.0	-24.8
		550	101.6	+17.9	78.1	+40.5	13.1	-43.8	30.4	-44.2
Weld Metal, (Wire Heat 25541 / Flux Lot 1211)	0.00	Room	79.4[b]	—	64.2[b]	—	19.2[b]	—	71.0[b]	—
		300	75.8[b]	—	62.0[b]	—	21.0[b]	—	68.0[b]	—
		550	78.7[b]	—	60.9[b]	—	19.0[b]	—	60.0[b]	—
	0.263	78	84.5	+6.4	70.8	+10.3	19.3	+0.5	65.0	-8.5
		548	84.5	+7.4	63.6	+4.4	19.0	0.0	56.5	-5.8
	0.872	275	86.6	+14.2	72.3	+16.6	16.5	-21.4	72	+5.9
		550	89.2	+13.3	71.8	+17.9	14.4	-24.2	66	+10.0
	2.052	200	86.8	+14.5[c]	73.0	+17.7[c]	16.0	-23.8[c]	60.8	-10.6[c]
		550	89.2	+13.3	72.9	+19.7	14.2	-25.3	48.6	-19.0

(a) Change relative to unirradiated material property.

(b) Mean value of available test data.

(c) Calculated relative to 300°F unirradiated tests.

Table 7-4. Measured vs. Predicted 30 ft-lb Transition Temperature Changes for North Anna Unit No. 1 Capsule W Surveillance Materials – 2.052 x 10^{19} n/cm^2

Material	Measured 30 ft-lb Transition Temperature, F			30 ft-lb Transition Temperature Shift Predicted in Accordance With Regulatory Guide 1.99, Revision 2				
	Unirradiated	Irradiated	Difference	Chemistry Factor[a]	ΔRT_{NDT}[b]	Margin (σ)	$\Delta RT_{NDT} - \sigma$	$\Delta RT_{NDT} + \sigma$
Base Metal Forging 03, Heat No. 990400/292332 (Tangential Orientation)	-5	88	93	120.0	143.5	17	126.5	160.5
Base Metal Forging 03, Heat No. 990400/292332 (Axial Orientation)	40	136	96	120.0	143.5	17	126.5	160.5
Weld Metal (Wire Heat 25531 / Flux Lot 1211)	-44	42	86	56.2	67.2	28	39.2	95.2
Heat-Affect-Zone Material	-76	8	84	120.0	143.5	17	126.5	160.5

(a) Chemistry factor based on mean copper and nickel contents as shown in Tables 7-1 and 7-2.

(b) ΔRT_{NDT} = Chemistry Factor * fluence factor (using the Capsule W fluence).

Table 7-5. Measured vs. Predicted Upper-Shelf Energy Decreases for North Anna Unit No. 1 Capsule W Surveillance Materials – 2.052 x 10^{19} n/cm^2

Material	Measured Upper-Shelf Energy, ft-lb			% Decrease Predicted In Accordance With Regulatory Guide 1.99, Rev. 2 Figure 2
	Unirradiated	Irradiated	% Decrease	
Base Metal Forging 03, Heat No. 990400/292332 (Tangential Orientation)	135	95	29.6	29.2[a]
Base Metal Forging 03, Heat No. 990400/292332 (Axial Orientation)	85	66	22.4	29.2[a]
Weld Metal (Wire Heat 25531 / Flux Lot 1211)	95	74	22.1	28.2[a]
Heat-Affect-Zone Material	146	89	39.0	29.2[a]

(a) Based on mean copper content as shown in Tables 7-1 and 7-2.

Table 7-6. Summary of North Anna Unit No. 1 Reactor Vessel Surveillance Capsules Charpy Impact Test Results

Material	Capsule	Fluence, 10^{19} n/cm^2	Measured Transition Temperature		Measured Upper-Shelf	
			ΔCv30, F	ΔCv50, F	Energy, ft-lb	% Decrease
Base Metal Forging 03, Heat No. 990400/292332 (Tangential Orientation)	Baseline	---	---	---	135	---
	V	0.263	51	61	122	9.6
	U	0.872	116	122	110	18.5
	W	2.052	93	114	95	29.6
Base Metal Forging 03, Heat No. 990400/292332 (Axial Orientation)	Baseline	---	---	---	85	---
	V	0.263	29	39	69	18.8
	U	0.872	72	81	93	-9.4
	W	2.052	96	122	66	22.4
Weld Metal (Wire Heat 25541 / Flux Lot 1211)	Baseline	---	---	---	95	---
	V	0.263	88	73	86	9.5
	U	0.872	30	78	92	3.2
	W	2.052	86	91	74	22.1
Heat-Affected Zone Material	Baseline	---	---	---	146	---
	V	0.263	57	74	103	29.5
	U	0.872	3	91	123	15.8
	W	2.052	84	107	89	39.0

Table 7-7. Summary of Sequoyah Unit No. 1 and Sequoyah Unit No. 2 Reactor Vessel Surveillance Capsules Charpy Impact Test Results[18]

Material	Capsule	Fluence, 10^{19} n/cm^2	Capsule Irradiation Temp., °F	Chemical Composition[20]		Measured ΔCv30, °F
				Cu wt%	Ni, wt%	
Sequoyah Unit No. 1 Surveillance Weld Metal (Wire Heat 25295)	T	0.288	545	0.375	0.125	128
	U	0.955	545	0.375	0.125	145
	X	1.39	545	0.375	0.125	157
Sequoyah Unit No. 2 Surveillance Weld Metal (Wire Heat 4278)	T	0.242	545	0.13	0.11	81
	U	0.608	545	0.13	0.11	154
	X	1.03	545	0.13	0.11	30

Table 7-8. Evaluation of Adjusted Reference Temperatures for the North Anna Unit No. 1 Reactor Vessel Applicable to 32.2 EFPY

Material Description[a]				Chemical Composition[a]				32.2 EFPY Fluence, n/cm^2			ΔRT_{NDT}, F at 32.2 EFPY		Margin		ART, F at 32.2 EFPY	
Reactor Vessel Beltline Region Location	Matl. Ident.	Heat Number	Type	Cu wt%	Ni wt%	Initial RT_{NDT}[a]	Chemistry Factor	Inside Surface	T/4 Location[b]	3/4T Location[b]	T/4 Location	3/4T Location	T/4 Location	3/4T Location	T/4 Location	3/4T Location
Regulatory Guide 1.99, Revision 2, Position 1.1																
Nozzle Belt Shell Forging	Forging 05	990286/ 295213	SA-508 Cl. 2	0.16	0.74	+6	121.5	3.885E+18	2.424E+18	9.436E+17	74.8	49.2	69.0	69.0	149.8	124.2
Intermediate Shell Forging	Forging 04	990311/ 298244	SA-508 Cl. 2	0.12	0.82	+17	86.0	4.108E+19	2.563E+19	9.978E+18	107.7	85.9	34.0	34.0	158.7	136.9
Lower Shell Forging	Forging 03	990400/ 292332	SA-508 Cl. 2	0.156	0.817	+38	120.0	4.108E+19	2.563E+19	9.978E+18	150.2	119.9	34.0	34.0	222.2	191.9
NS to IS Circ. Weld (OD 94%)	W05A	25295	ASA/SMIT 89	0.352	0.125	0	163.3	N/A	2.424E+18	9.436E+17	100.3	65.9	68.8	68.8	169.1	134.7
NS to IS Circ. Weld (ID 6%)	W05B	4278	ASA/SMIT 89	0.12	0.11	0	63.0	3.885E+18	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A
IS to LS Circ. Weld (100%)	W04	25531	ASA/SMIT 89	0.098	0.124	+19	56.2	4.108E+19	2.563E+19	9.978E+18	70.4	56.1	56.0	56.0	145.4	131.1
Regulatory Guide 1.99, Revision 2, Position 2.1																
Lower Shell Forging	Forging 03	990400/ 292332	SA-508 Cl. 2	0.156	0.817	+38	82.9	4.108E+19	2.563E+19	9.978E+18	103.8	82.8	34.0[c]	34.0[c]	[175.8]	[154.8]
NS to IS Circ. Weld (OD 94%)	W05A	25295	ASA/SMIT 89	0.352	0.125	0	137.7	N/A	2.424E+18	9.436E+17	84.8	55.8	48.8	48.8	133.6	104.6
NS to IS Circ. Weld (ID 6%)	W05B	4278	ASA/SMIT 89	0.12	0.11	0	85.2	3.885E+18	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A
IS to LS Circ. Weld (100%)	W04	25531	ASA/SMIT 89	0.098	0.124	+19	68.0	4.108E+19	2.563E+19	9.978E+18	85.1	67.9	56.0[c]	56.0[c]	160.1	142.9

[a] See Appendix A.

[b] Calculated based on the guidelines in Regulatory Guide 1.99, Revision 2 ("x" for ¼T = 1.9655 in. and "x" for ¾T = 5.8965 in.).

[c] Two of the surveillance data points are not credible, however, all surveillance data points are conservatively bounded by the $+2\sigma$ curve based on Regulatory Guide 1.99, Revision 2, Position 1.1 chemistry factor. Therefore, a full margin value and the Regulatory Guide 1.99, Revision 2, Position 2.1 chemistry factor are used to calculate the adjusted reference temperature value.

[ ] Controlling values of the adjusted reference temperatures.

Table 7-9. Evaluation of Pressurized Thermal Shock Reference Temperatures for the North Anna Unit No. 1 Reactor Vessel Applicable to 32.2 EFPY

Material Description[a]				Chemical Composition[a]									
Reactor Vessel Beltline Region Matl.	Matl. Ident.	Heat Number	Type	Cu wt%	Ni wt%	Chem. Factor	Initial RT_{NDT}, F[a]	32.2 EFPY Fluence, n/cm^2	Fluence Factor	ΔRT_{PTS}, F	Margin, F	RT_{PTS}, F	Screening Criteria
RT_{PTS} Calculation Per 10 CFR 50.61 Using Tables													
Nozzle Belt Shell Forging (NS)	Forging 05	990286/ 295213	SA-508 Cl. 2	0.16	0.74	121.5	6	3.885E+18	0.738	89.7	69.0	164.7	270
Intermediate Shell Forging (IS)	Forging 04	990311/ 298244	SA-508 Cl. 2	0.12	0.82	86.0	17	4.108E+19	1.362	117.1	34.0	168.1	270
Lower Shell Forging (LS)	Forging 03	990400/ 292332	SA-508 Cl. 2	0.156	0.817	120.0	38	4.108E+19	1.362	163.4	34.0	235.4	270
NS to IS Circ. Weld (OD 94%)	W05A	25295	ASA/SMIT 89	0.352	0.125	163.3	0	N/A	N/A	N/A	N/A	N/A	300
NS to IS Circ. Weld (ID 6%)	W05B	4278	ASA/SMIT 89	0.12	0.11	63.0	0	3.885E+18	0.738	46.5	61.3	107.8	300
IS to LS Circ. Weld (100%)	W04	25531	ASA/SMIT 89	0.098	0.124	56.2	19	4.108E+19	1.362	76.5	56.0	151.5	300
RT_{PTS} Calculation Per 10 CFR 50.61 Using Surveillance Data													
Lower Shell Forging (LS)	Forging 03	990400/ 292332	SA-508 Cl. 2	0.156	0.817	82.9	38	4.108E+19	1.362	112.9	34.0[b]	[184.9]	270
NS to IS Circ. Weld (OD 94%)	W05A	25295	ASA/SMIT 89	0.352	0.125	137.7	0	N/A	N/A	N/A	N/A	N/A	300
NS to IS Circ. Weld (ID 6%)	W05B	4278	ASA/SMIT 89	0.12	0.11	85.2	0	3.885E+18	0.738	62.9	68.8[b]	131.7	300
IS to LS Circ. Weld (100%)	W04	25531	ASA/SMIT 89	0.098	0.124	68.0	19	4.108E+19	1.362	92.6	56.0[b]	167.6	300

[a] See Appendix A.

[b] Two of the surveillance data points are not credible, however, all surveillance data points are conservatively bounded by the $+2\sigma$ curve based on generic chemistry factor Tables in 10 CFR 50.61. Therefore, a full margin value and the chemistry factor determined using surveillance data are used to calculate the adjusted reference temperature value.

[ ] Limiting reactor vessel beltline region material in accordance with 10 CFR 50.61.

8. Summary of Results

The analysis of the reactor vessel material contained in the third surveillance capsule, Capsule W, removed for evaluation as part of the North Anna Unit No. 1 Reactor Vessel Surveillance Program, led to the following conclusions:

1. The capsule received an average fast neutron fluence of 2.052×10^{19} n/cm^2 (E > 1.0 MeV).

2. Based on the calculated cycle 11, 12, and 13 full power flux weighted average, the projected end-of-life (32.2 EFPY) peak fast fluence of the North Anna Unit No. 1 reactor vessel beltline region is 4.108×10^{19} n/cm^2 (E > 1.0 MeV). The corresponding fluences based on the FTI fluence methodology at the ¼-thickness and ¾-thickness vessel wall locations in this peak location are 2.576×10^{19} and 5.486×10^{18} n/cm^2 (E > 1.0 MeV) respectively.

3. The 30 ft-lb transition temperature for the surveillance base metal forging (Forging 03), heat no. 990400/292332, in the tangential orientation, increased 93°F after the irradiation to 2.052×10^{19} n/cm^2 (E > 1.0 MeV). In addition, the C_vUSE for this material decreased 29.6%.

4. The 30 ft-lb transition temperature for the surveillance base metal forging (Forging 03), heat no. 990400/292332, in the axial orientation, increased 96°F after the irradiation to 2.052×10^{19} n/cm^2 (E > 1.0 MeV). In addition, the C_vUSE for this material decreased 22.4%.

5. The 30 ft-lb transition temperature for the surveillance weld metal, weld wire heat 25531 / flux lot 1211, increased 86°F after the irradiation to 2.052×10^{19} n/cm^2 (E > 1.0 MeV). In addition, the C_vUSE for this material decreased 22.1%.

6. The measured upper-shelf energies for the North Anna Unit No. 1 Capsule W surveillance materials do not fall below the required 50 ft-lbs limit after the irradiation to 2.052×10^{19} n/cm^2 (E > 1.0 MeV).

7. In accordance with 10 CFR 50.61, the North Anna Unit No. 1 reactor vessel beltline materials will not exceed the PTS screening criteria before end-of-life (32.2 EFPY).

9. Certification

The specimens obtained from the Virginia Power North Anna Unit No. 1 reactor vessel surveillance capsule (Capsule W) were tested and evaluated using accepted techniques and established standard methods and procedures in accordance with the requirements of 10 CFR 50, Appendices G and H.

9/1/99

M. J. DeVan (Material Analysis) Date
Materials & Structural Analysis Unit

9/1/99

E. Giavedoni (Fluence Analysis) Date
Performance Analysis Unit

This report has been reviewed for technical content and accuracy.

9-2-99

J. B. Hall (Material Analysis) Date
Materials & Structural Analysis Unit

9-2-99

S. Q. King (Fluence Analysis) Date
Performance Analysis Unit

Verification of independent review.

9-2-99

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

This report is approved for release.

9/2/99

D. L. Howell Date
Program Manager

Revision 1

The revisions to this report were made as stated in accordance with the standard methods and procedures for the original report.

M. J. DeVan 11/5/99

M. J. DeVan (Material Analysis) Date
Materials & Structural Analysis Unit

11/5/99

E. Giavedoni (Fluence Analysis) Date
Performance Analysis Unit

This report has been reviewed for technical content and accuracy.

JB Hall 11-5-99

J. B. Hall (Material Analysis) Date
Materials & Structural Analysis Unit 1

for S.Q. King 11/5/99

S. Q. King (Fluence Analysis) Date
Performance Analysis Unit

Verification of independent review.

JFShepard for KE Moore 11/5/99

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

This report is approved for release.

DLHowell 11/5/99

D. L. Howell Date
Program Manager

10. References

1. 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.

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

3. S. E. Yanichko, L. Albertin, and E. P. Lippincott, *"Analysis of Capsule U from the Virginia Electric & Power Company North Anna Unit 1 Reactor Vessel Radiation Surveillance Program,"* WCAP-11777, Westinghouse Electric Corporation, Pittsburgh, Pennsylvania, February 1988.

4. ASTM Standard E 185-73, *"Standard Recommended Practice for Surveillance Tests for Nuclear Reactor Vessels,"* American Society for Testing and Materials, Philadelphia, Pennsylvania.

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

6. Code of Federal Regulation, Title 10, Part 50, *"Domestic Licensing of Production and Utilization Facilities,"* Appendix H, Reactor Vessel Material Surveillance Program Requirements.

7. American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section III, *"Nuclear Power Plant Components,"* Appendix G, Protection Against Nonductile Failure, 1989 Edition.

8. American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section XI, *"Rules for Inservice Inspection of Nuclear Power Plant Components,"* Appendix G, Fracture Toughness Criteria for Protection Against Failure, 1989 Edition.

9. ASTM Standard E 208-81, *"Method for Conducting Drop-Weight Test to Determine Nil-Ductility Transition Temperature of Ferritic Steels,"* American Society for Testing and Materials, Philadelphia, Pennsylvania.

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* - Available from Framatome Technologies Inc., Lynchburg, Virginia.

10. K. Y. Hour, *"Evaluation of Virginia Power Company North Anna Unit 1 Capsule W,"* 00:475-0189-01:02 (FTG Document No. 31-1023292-01), B&W Services, Inc., Lynchburg, Virginia, August 1999.

11. ASTM Standard E 21-92, *"Standard Test Methods for Elevated Temperature Tension Tests of Metallic Materials,"* American Society for Testing and Materials, Philadelphia, Pennsylvania.

12. ASTM Standard E 23-91, *"Standard Test Methods for Notched Bar Impact Testing of Metallic Materials,"* American Society for Testing and Materials, Philadelphia, Pennsylvania.

13. ASTM Standard E 185-82, *"Standard Practice for Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels, E 706 (IF)"* American Society for Testing and Materials, Philadelphia, Pennsylvania.

14. J. R. Worsham III, *"Fluence and Uncertainity Methodologies,"* BAW-2241P, Revision 1, Framatome Technologies, Inc., Lynchburg, Virginia, April 1999.

15. U.S. Nuclear Regulatory Commission, *"Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence,"* Draft Regulatory Guide DB-1053, June 1996.

16. E. P. Lippincott, *"Westinghouse Surveillance Capsule Neutron Fluence Reevaluation,"* WCAP-14044, Westinghouse Electric Corporation, Pittsburgh, Pennsylvania, April 1994.

17. U.S. Nuclear Regulatory Commission, *"Radiation Embrittlement of Reactor Vessel Materials,"* Regulatory Guide 1.99, Revision 2, May 1998.

18. E. D. Eason, J. E. Wright, and G. R. Odette, *"Improved Embrittlement Correlations for Reactor Pressure Vessel Steels,"* NUREG/CR-6551, Modeling and Computing Services, Boulder, Colorado, August 1998.

19. Code of Federal Regulations, Title 10, *"Domestic Licensing of Production and Utilization Facilities,"* Part 50.61, *"Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock,"* Federal Register, dated December 19, 1995 as amended by Federal Register, dated July 29, 1996.

20. M. J. DeVan, *"Response to Generic Letter 92-01, Revision 1, Supplement 1 for Virginia Power' North Anna Units 1 and 2 Beltline Materials and Surry Units 1 and 2 Rotterdam Beltline Weld Metals,"* BAW-2260, B&W Nuclear Technologies, Inc., Lynchburg, Virginia, October 1995.*

* - Available from Framatome Technologies Inc., Lynchburg, Virginia.

APPENDIX A

Reactor Vessel Surveillance Program Background Data and Information

A.1. Capsule Identification

The capsules, used in the North Anna Unit No. 1 reactor vessel surveillance program, are identified in Table A-1 by identification and location. The capsule locations within the North Anna Unit No. 1 reactor vessel are shown in Figure A-2.

A.2. North Anna Unit No. 1 Reactor Pressure Vessel

The North Anna Unit No. 1 reactor pressure vessel was fabricated by the Rotterdam Dockyard Company. The North Anna Unit No. 1 reactor vessel beltline region consists of two shells, containing two heats of base metal forging and one circumferential weld seam. Table A-2 presents a description of the North Anna Unit No. 1 reactor vessel beltline materials including their copper and nickel chemical contents and their unirradiated mechanical properties. The heat treatments of the beltline materials are presented in Table A-3. The locations of the materials within the reactor vessel beltline region are shown in Figure A-1.

A.3. Surveillance Material Selection Data

The data used to select the materials for the specimens in the surveillance program, in accordance with ASTM Standard E 185-73, are shown in Table A-2. The North Anna Unit No. 1 RVSP capsules include the limiting reactor vessel beltline material, Forging 03, heat no. 990400/292332. The surveillance weld used in the North Anna Unit No. 1 RVSP was fabricated using the wire heat 25531 and SMIT 89 flux lot 1211 which is identical to the intermediate to lower shell circumferential weld in the North Anna Unit No. 1 reactor vessel.

Table A-1. North Anna Unit No. 1 Surveillance Capsule Identifications and Original Locations

Capsule Identification	Capsule Location[a]
S	45°
T	55°
U	65°
V	165°
W	245°
X	285°
Y	295°
Z	305°

(a) Reference irradiation capsule locations as shown in Figure A-2.

Table A-2. Description of the North Anna Unit No. 1 Reactor Vessel Beltline Region Materials[A-1],[A-2],[A-3],[A-4]

Material Heat No.	Material Type	Beltline Region Location	Chemical Composition		Unirradiated Toughness Properties					
			Cu, wt%	Ni, wt%	30 ft-lb, F	50 ft-lb, F	35 MLE, F	C_VUSE, ft-lbs	T_{NDT}, F	RT_{NDT}, F
990286/295213	SA-508 Cl. 2	Nozzle Belt Shell	0.16	0.74	---	---	---	74	2	6
990311/298244	SA-508 Cl. 2	Intermediate Shell	0.12	0.82	---	---	---	92	-31	17
990400/292332	SA-508 Cl. 2	Lower Shell	0.156(b)	0.817(b)	---	---	---	85	-13	38
25295 / 1170(a)	ASA Weld/ SMIT 89 Flux	Nozzle Belt to Interm. Shell Circ. Weld (OD 94%)	0.352	0.125	---	---	---	111	0	0
4278 / 1211(a)	ASA Weld/ SMIT 89 Flux	Nozzle Belt to Interm. Shell Circ. Weld (ID 6%)	0.12	0.11	---	---	---	105	0	0
25531 / 1211(a)	ASA Weld/ SMIT 89 Flux	Intermediate to Lower Shell Circ. Weld	0.098(b)	0.124(b)	---	---	---	102	-13	19

(a) Weld wire heat number and flux lot identifiers.
(b) New best estimate values (see Section 7).

Table A-3. Heat Treatment of the North Anna Unit No. 1 Reactor Vessel Beltline Region Materials

Material	Heat Treatment
Nozzle Belt Forging 05 (Ht. No. 990286/295213)	Austenitizing: 1616-1697°F for 3 hrs., water quenched Tempering: 1202-1238°F for 6 hrs., furnace cooled to 761°F Post Weld: 1130±25°F for 14¾ hrs. (min.), furnace cooled[(a),(b)]
Intermediate Shell Forging 04 (Ht. No. 990311/29824)	Austenitizing: 1616-1697°F for 6 hrs., water quenched Tempering: 1202-1238°F for 6 hrs., furnace cooled to 824°F Post Weld: 1130±25°F for 14¾ hrs. (min.), furnace cooled[(a),(b)]
Lower Shell Forging 03 (Ht. No. 990400/292332)	Austenitizing: 1616-1706°F for 5 hrs., water quenched Tempering: 1202-1247°F for 7½ hrs., furnace cooled to 851°F Post Weld: 1130±25°F for 14¾ hrs. (min.), furnace cooled[(a),(b)]
Nozzle Belt to Intermediate Shell Girth Seam Weld (OD 94%) (Wire Heat 25295/Flux Lot 1170)	Post Weld: 1130±25°F for 10¾ hrs. (min.), furnace cooled[(c)]
Nozzle Belt to Intermediate Shell Girth Seam Weld (ID 6%) (Wire Heat 4278/Flux Lot 1211)	Post Weld: 1130±25°F for 10¾ hrs. (min.), furnace cooled[(c)]
Intermediate to Lower Shell Girth Seam Weld (Wire Heat 25531/Flux Lot 1211)	Post Weld: 1130±25°F for 10¾ hrs. (min.), furnace cooled[(c)]

(a) Austenitizing and tempering times are from Rotterdam Dockyard Company Test Certificates.[A-5]

(b) Post weld heat treatments based on heat treatment of North Anna Unit No. 1 surveillance base metal.

(c) Post weld heat treatments based on heat treatment of North Anna Unit No. 1 surveillance weld metal.

Figure A-1. Location and Identification of Materials Used in the Fabrication of the North Anna Unit No. 1 Reactor Pressure Vessel

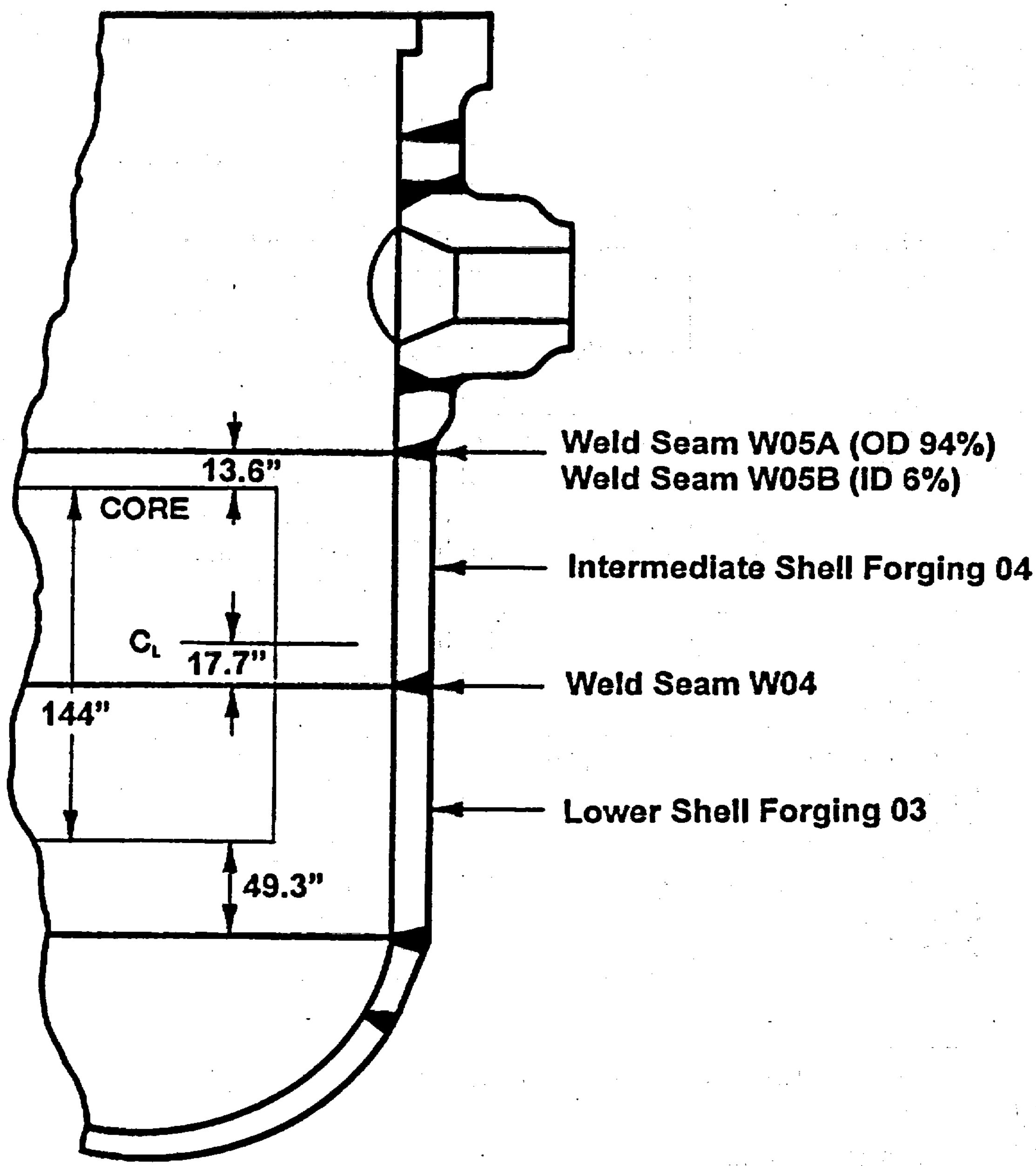
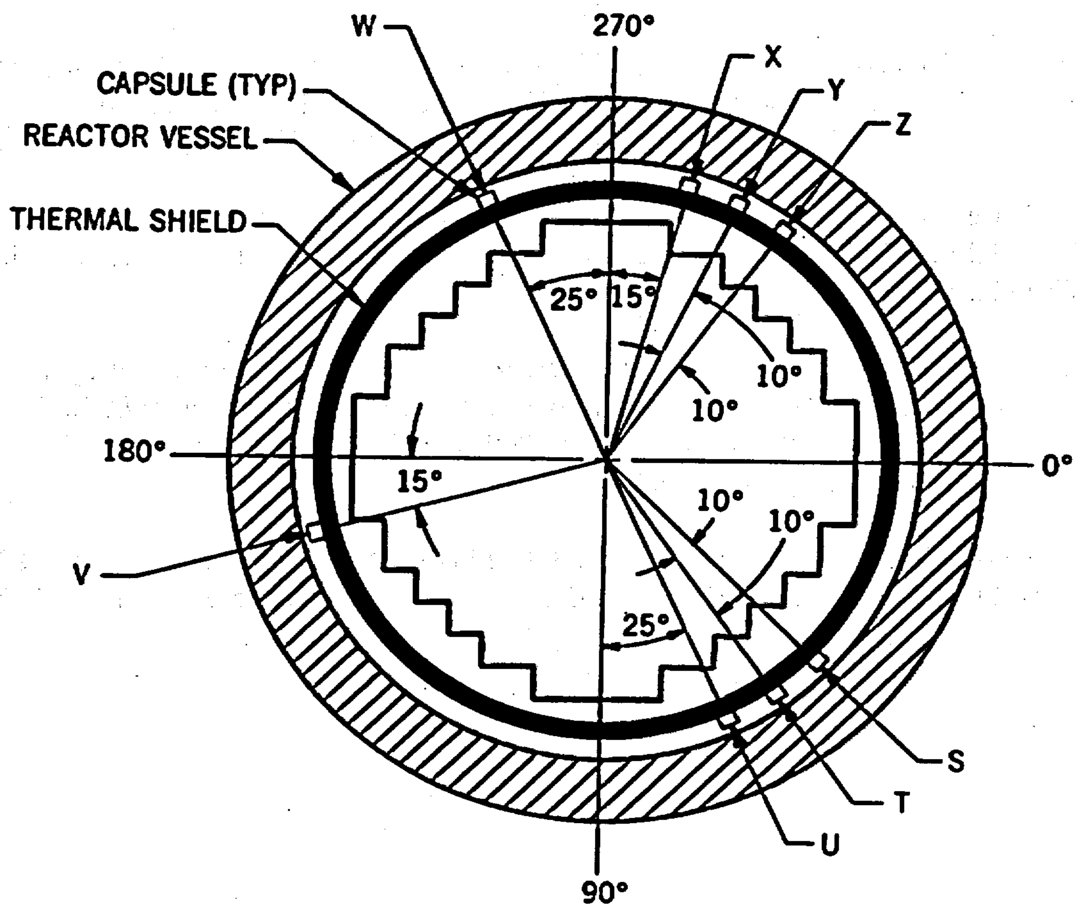


Figure A-2. Original Locations of Surveillance Capsule Irradiation Sites in the North Anna Unit No. 1 Reactor Vessel



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A.4. References

A-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.*

A-2. M. J. DeVan and A. L. Lowe, Jr., *"Response to Generic Letter 92-01 for Virginia Electric & Power Company North Anna Unit 1 and North Anna Unit 2,* BAW-2168, Revision 1, B&W Nuclear Technologies, Inc., Lynchburg, Virginia, September 1992.*

A-3. M. J. DeVan, *"North Anna Units 1 and 2 Response to Closure Letter for NRC Generic Letter 92-01, Revision 1,"* BAW-2224, B&W Nuclear Technologies, Inc., Lynchburg, Virginia, July 1994.*

A-4. M. J. DeVan, *"Response to Generic Letter 92-01, Revision 1, Supplement 1 for Virginia Power' North Anna Units 1 and 2 Beltline Materials and Surry Units 1 and 2 Rotterdam Beltline Weld Metals,"* BAW-2260, B&W Nuclear Technologies, Inc., Lynchburg, Virginia, October 1995.*

A-5. Framatome Technologies Inc. Document 38-1247870-00, *"North Anna Units 1 &2 and Surry Units 1 & 2 Reactor Vessel Beltline Materials Data,"* release September 1999.

* - Available from Framatome Technologies Inc., Lynchburg, Virginia.

APPENDIX B

Instrumented Charpy V-Notch Specimen Test Results
Load-Time Traces

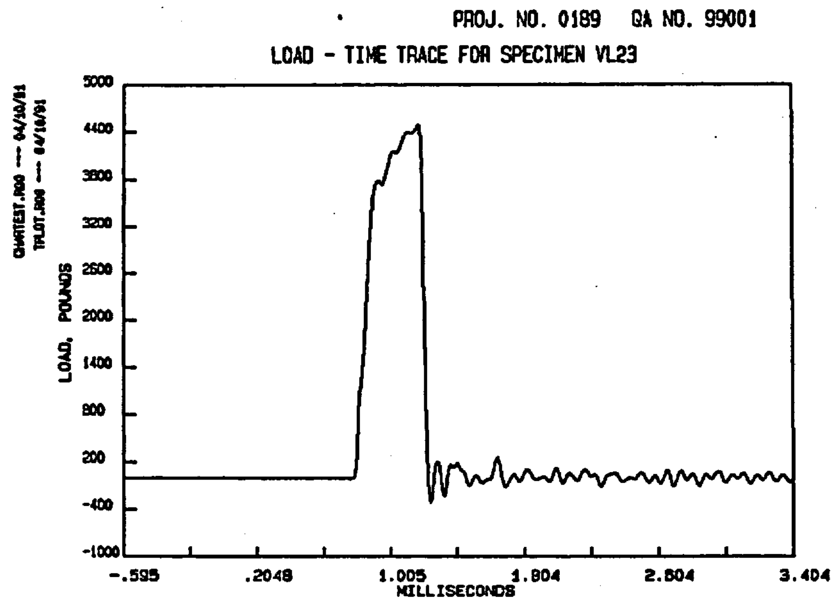

Figure B-1. Load-Time Trace for Charpy V-Notch Impact Specimen VL23

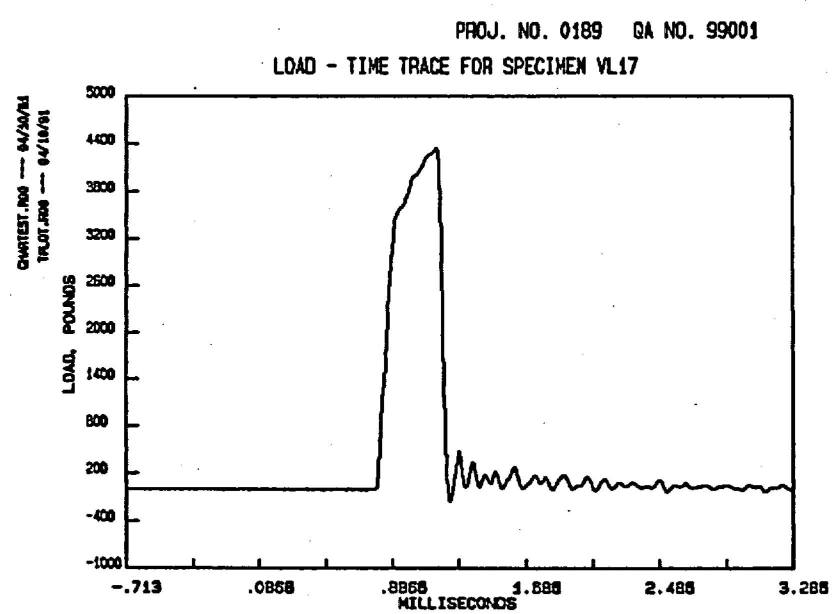

Figure B-2. Load-Time Trace for Charpy V-Notch Impact Specimen VL17

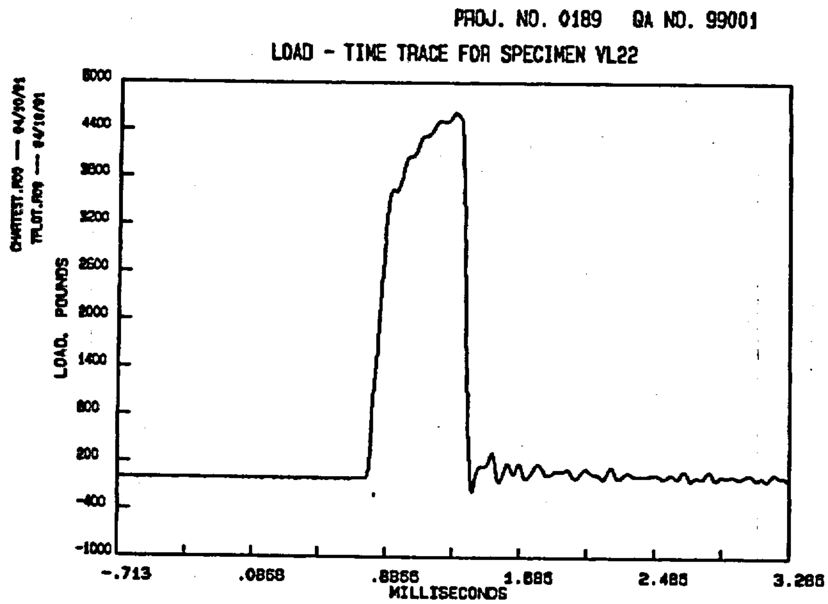

Figure B-3. Load-Time Trace for Charpy V-Notch Impact Specimen VL22

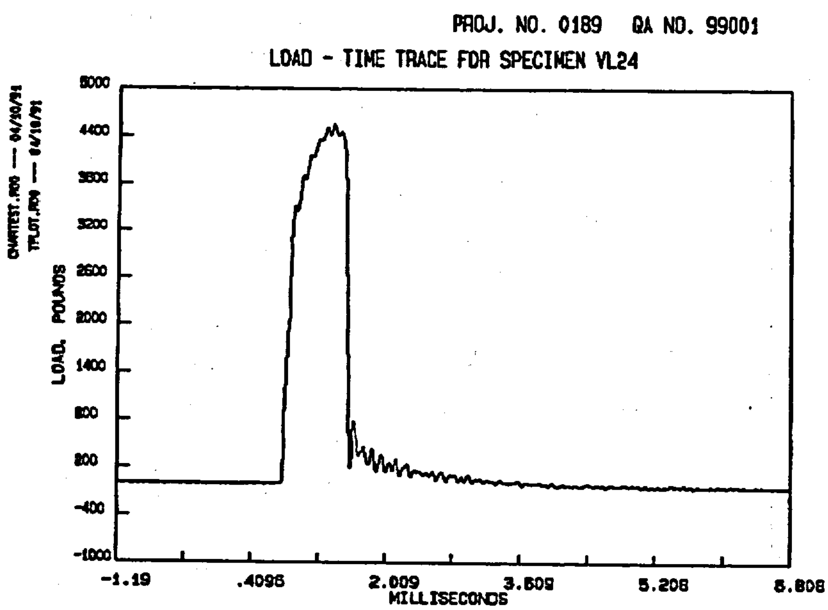

Figure B-4. Load-Time Trace for Charpy V-Notch Impact Specimen VL24

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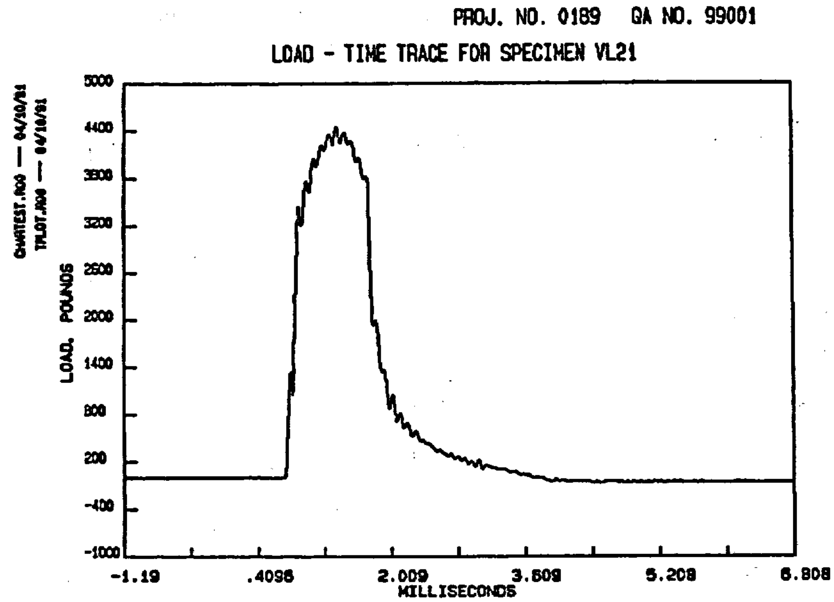

Figure B-5. Load-Time Trace for Charpy V-Notch Impact Specimen VL21

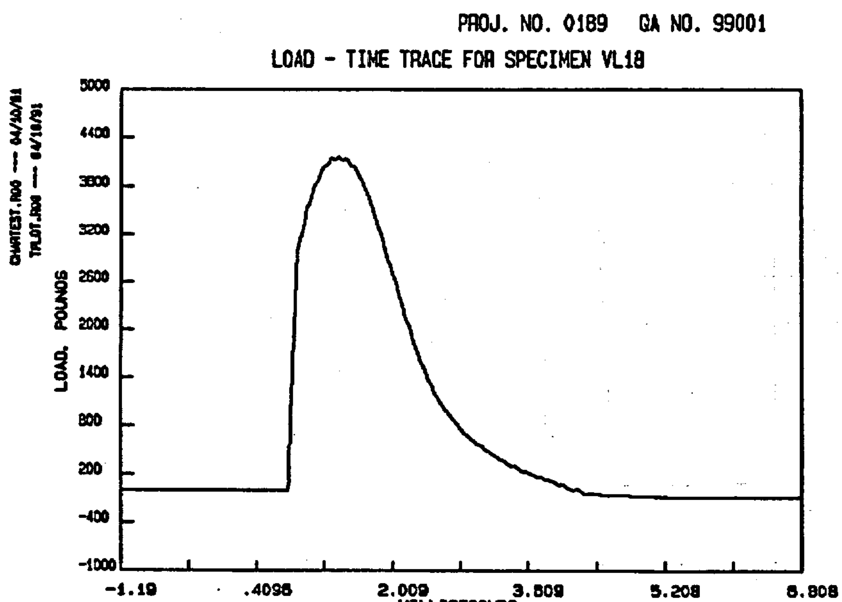

Figure B-6. Load-Time Trace for Charpy V-Notch Impact Specimen VL18

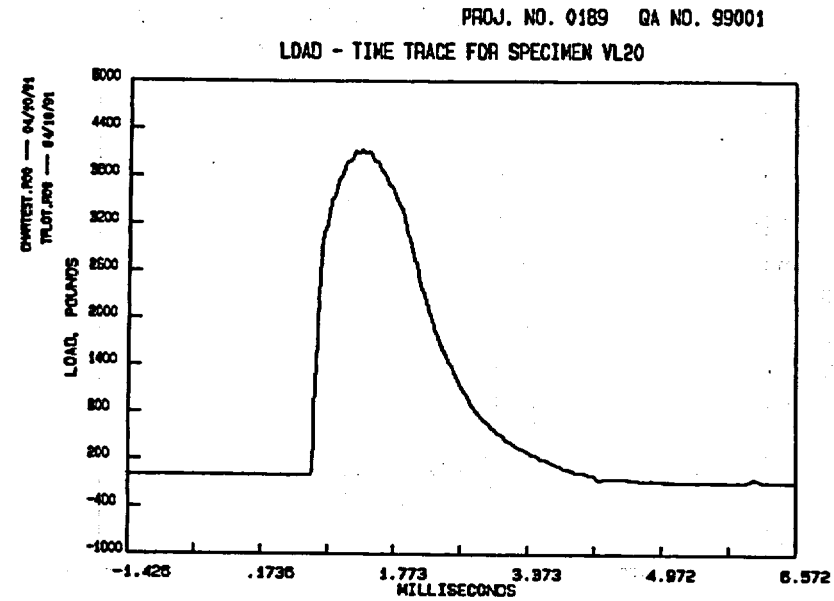

Figure B-7. Load-Time Trace for Charpy V-Notch Impact Specimen VL20

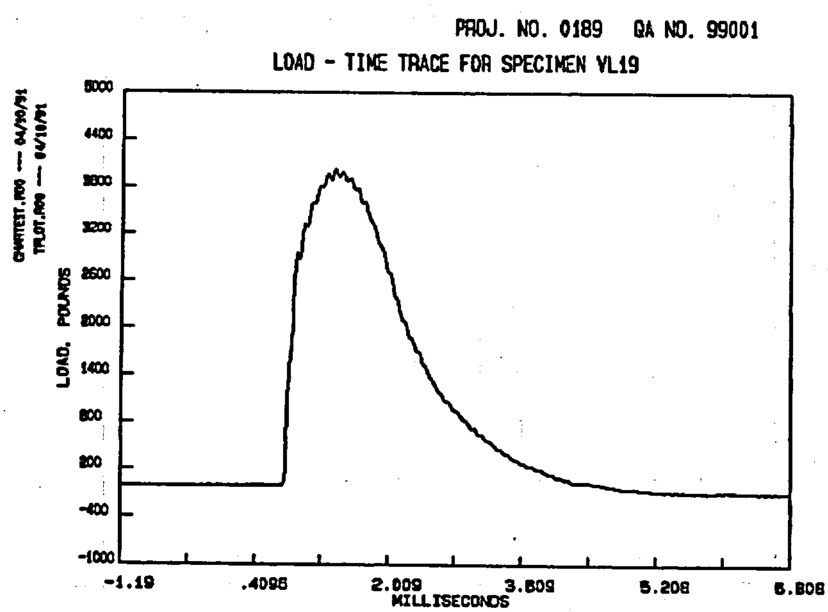

Figure B-8. Load-Time Trace for Charpy V-Notch Impact Specimen VL19

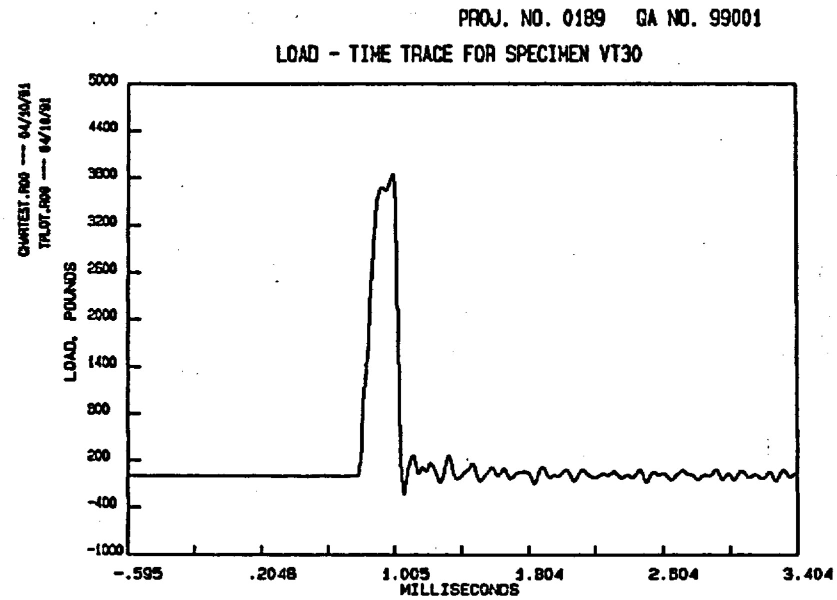

Figure B-9. Load-Time Trace for Charpy V-Notch Impact Specimen VT30

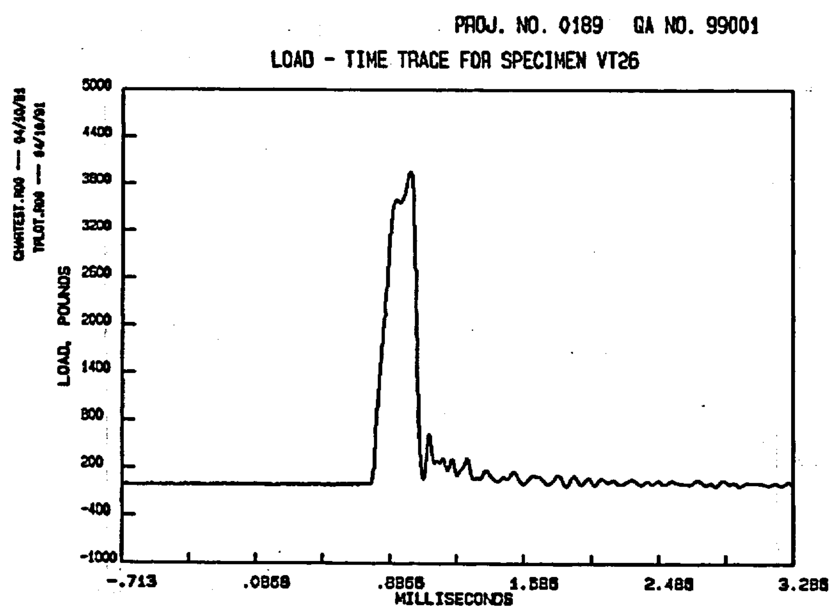

Figure B-10. Load-Time Trace for Charpy V-Notch Impact Specimen VT26

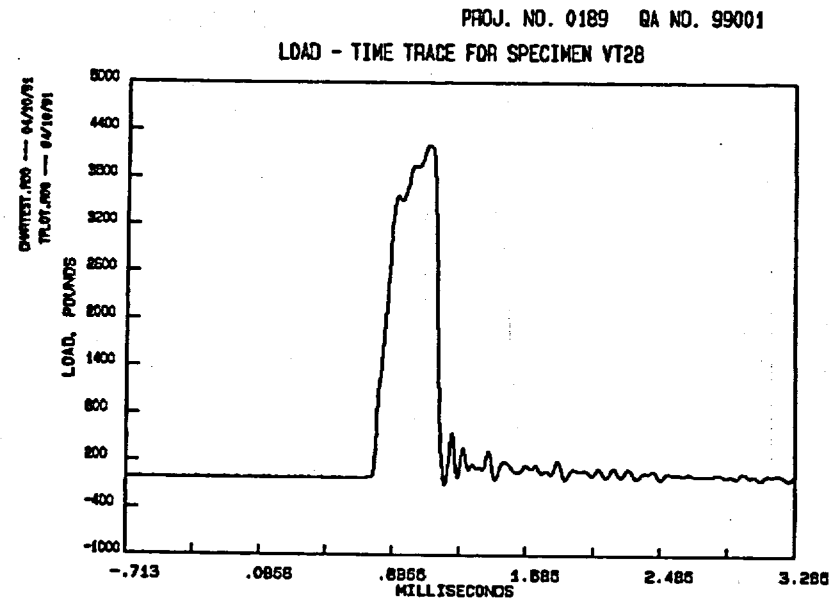

Figure B-11. Load-Time Trace for Charpy V-Notch Impact Specimen VT28

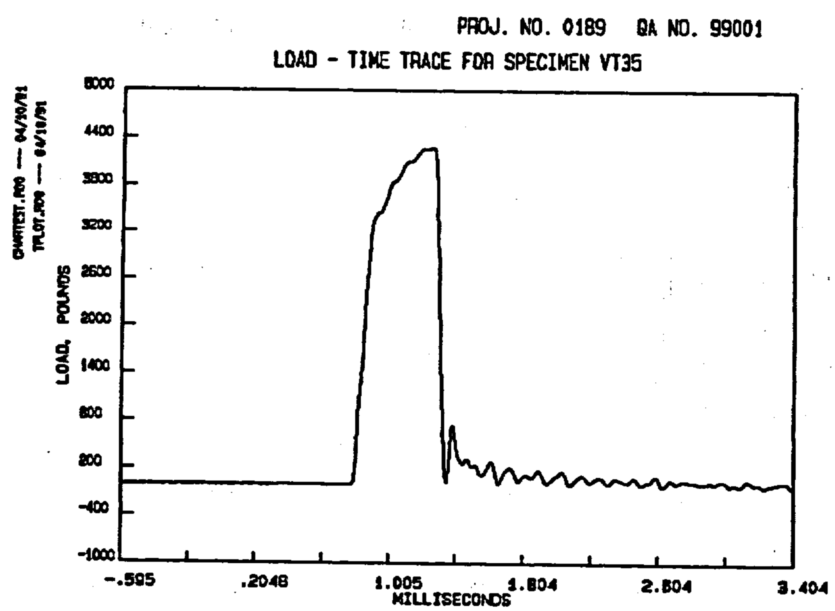

Figure B-12. Load-Time Trace for Charpy V-Notch Impact Specimen VT35

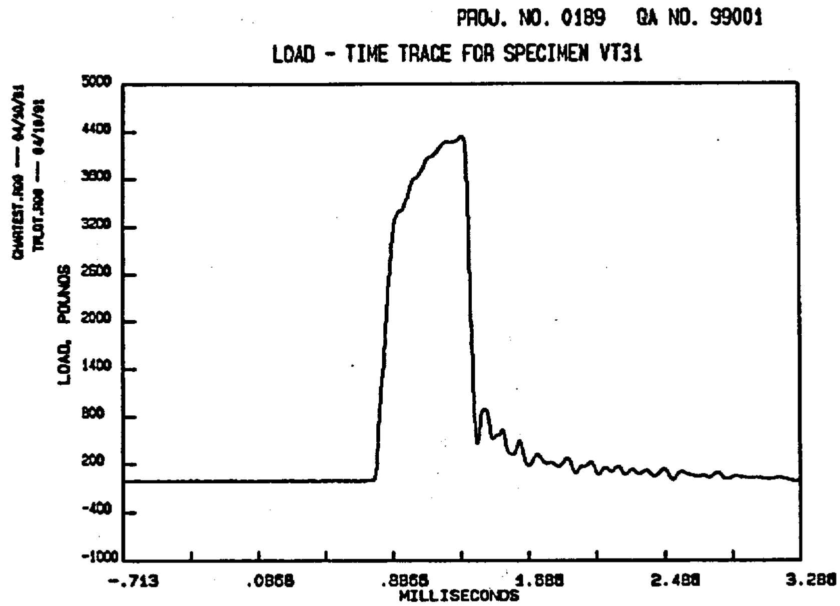

Figure B-13. Load-Time Trace for Charpy V-Notch Impact Specimen VT31

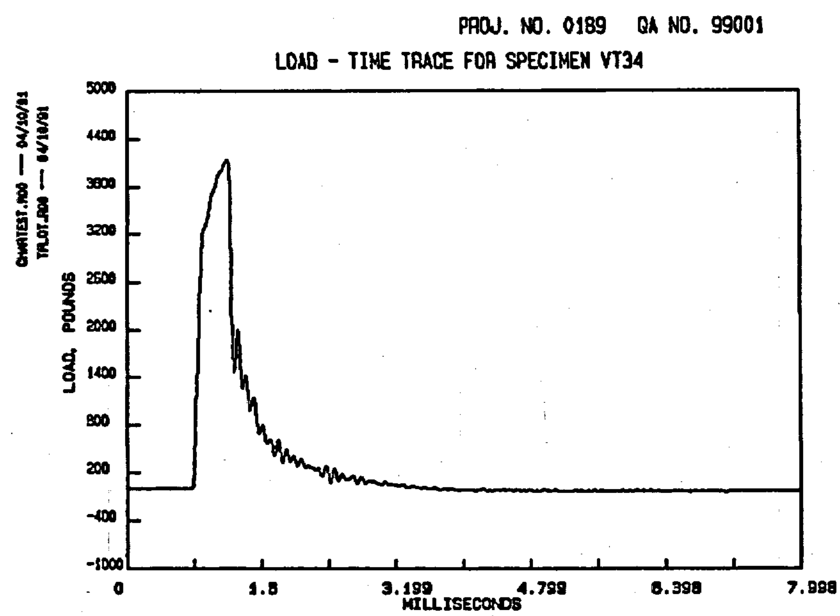

Figure B-14. Load-Time Trace for Charpy V-Notch Impact Specimen VT34

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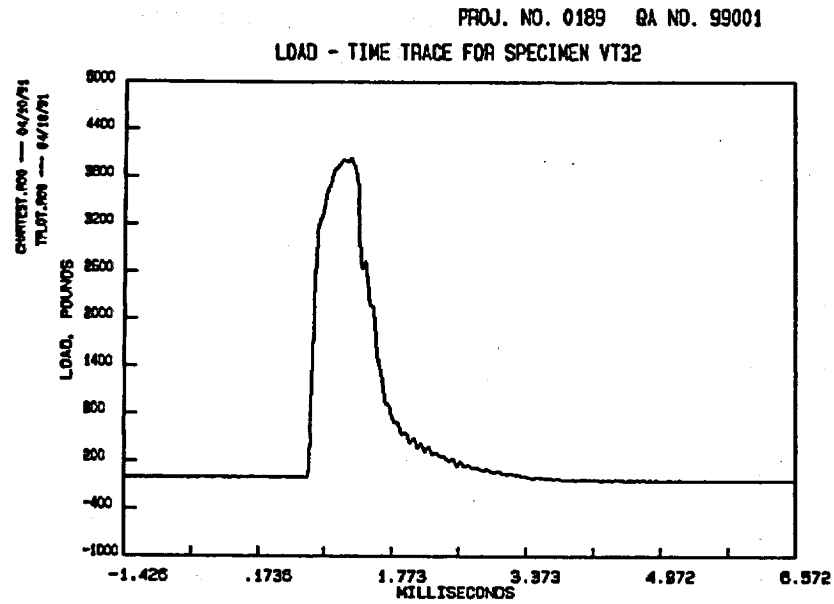

Figure B-15. Load-Time Trace for Charpy V-Notch Impact Specimen VT32

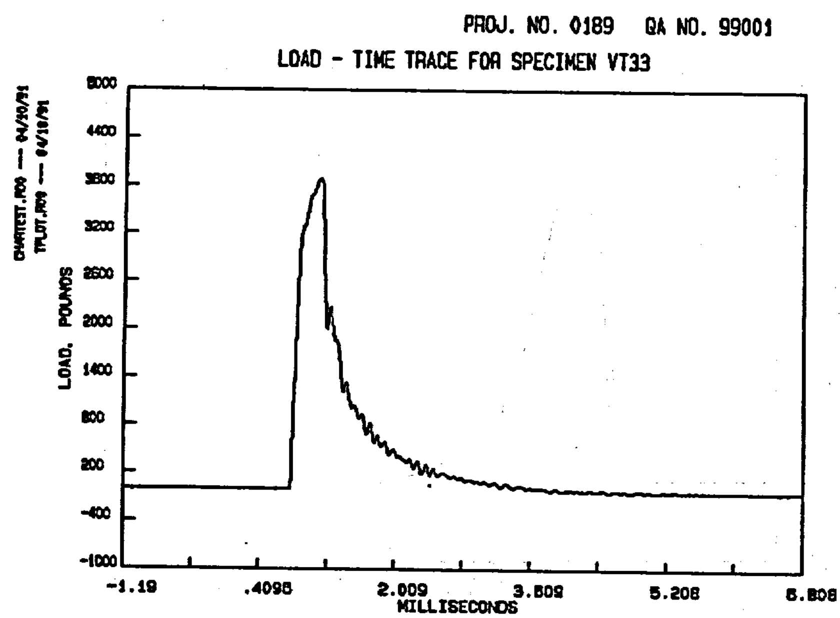

Figure B-16. Load-Time Trace for Charpy V-Notch Impact Specimen VT33

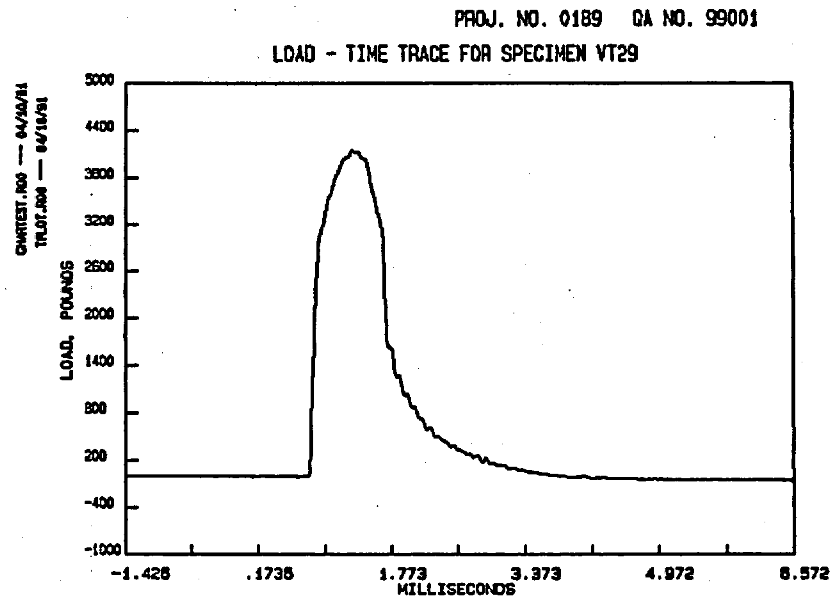

Figure B-17. Load-Time Trace for Charpy V-Notch Impact Specimen VT29

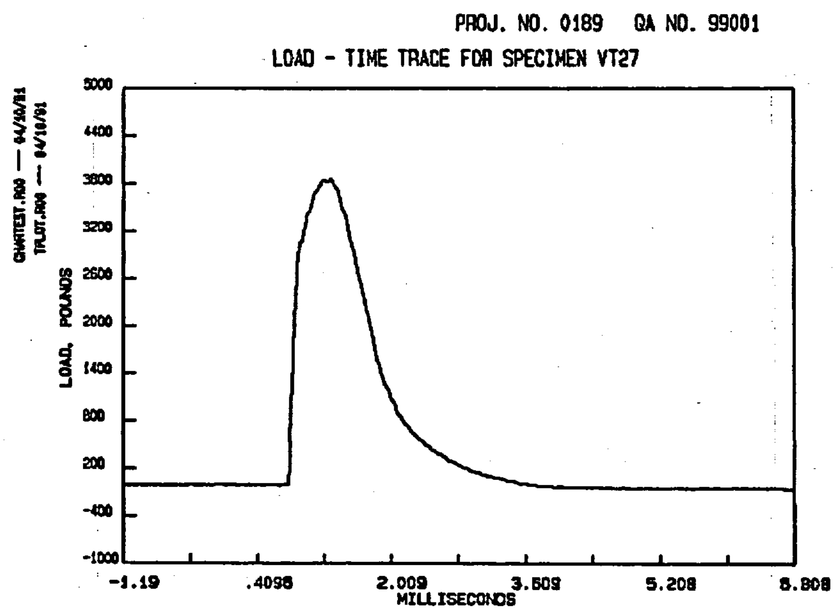

Figure B-18. Load-Time Trace for Charpy V-Notch Impact Specimen VT27

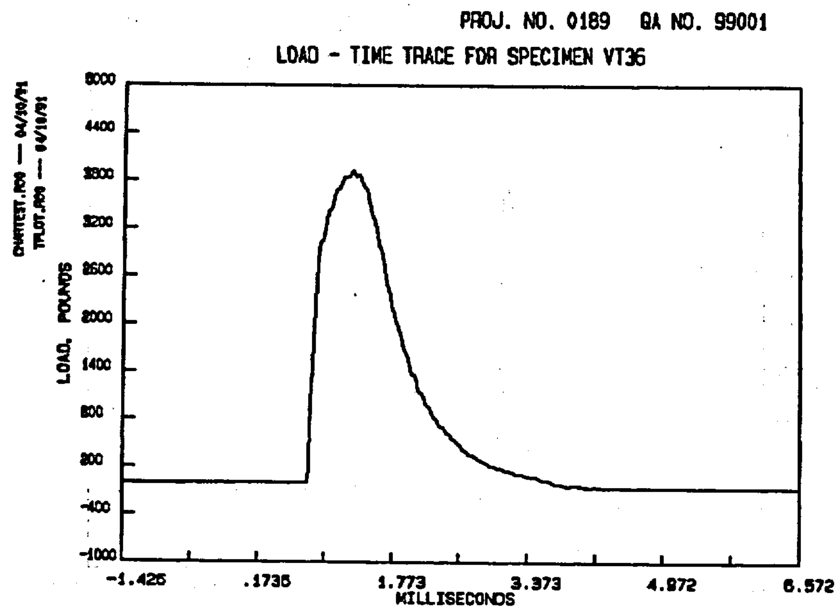

Figure B-19. Load-Time Trace for Charpy V-Notch Impact Specimen VT36

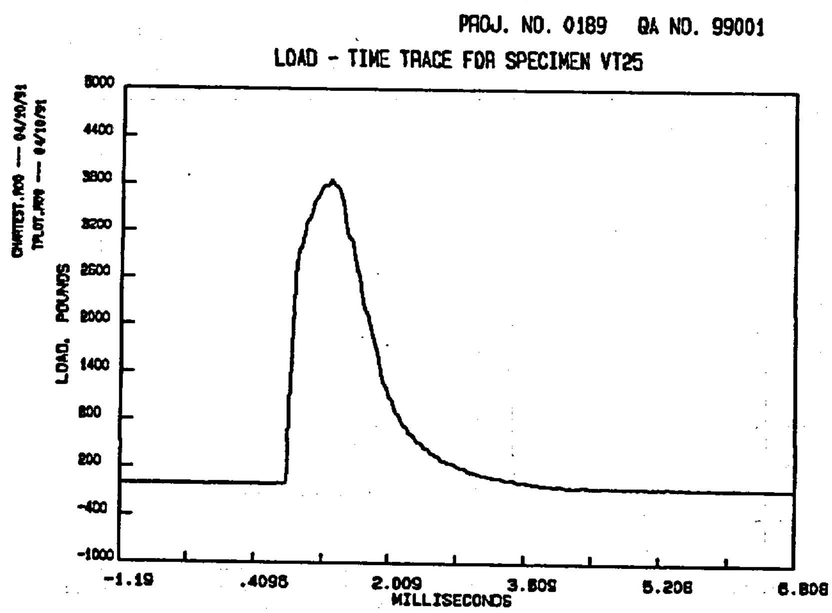

Figure B-20. Load-Time Trace for Charpy V-Notch Impact Specimen VT25

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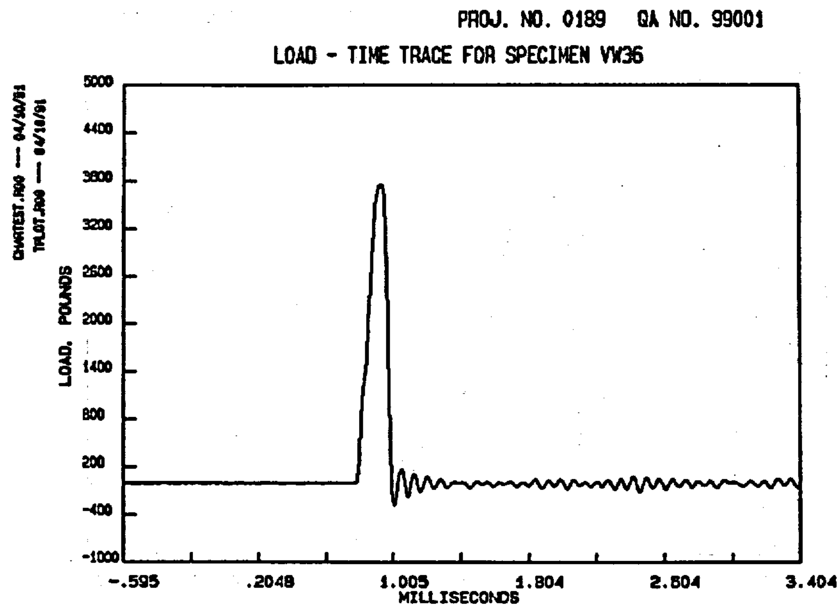

Figure B-21. Load-Time Trace for Charpy V-Notch Impact Specimen VW36

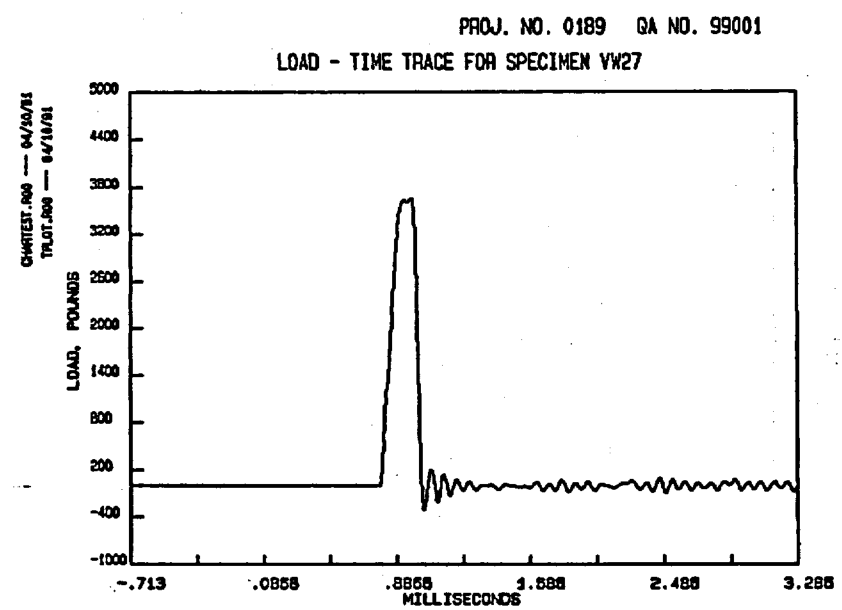

Figure B-22. Load-Time Trace for Charpy V-Notch Impact Specimen VW27

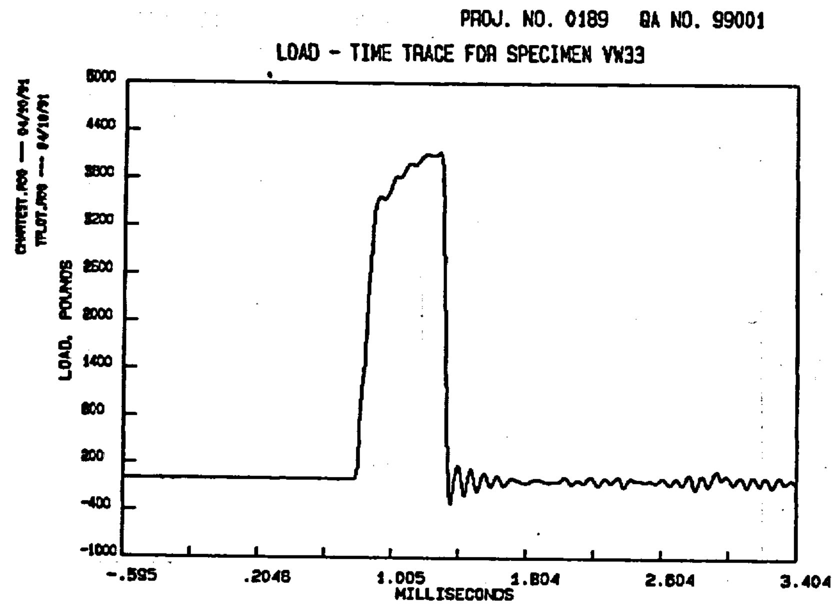

Figure B-23. Load-Time Trace for Charpy V-Notch Impact Specimen VW33

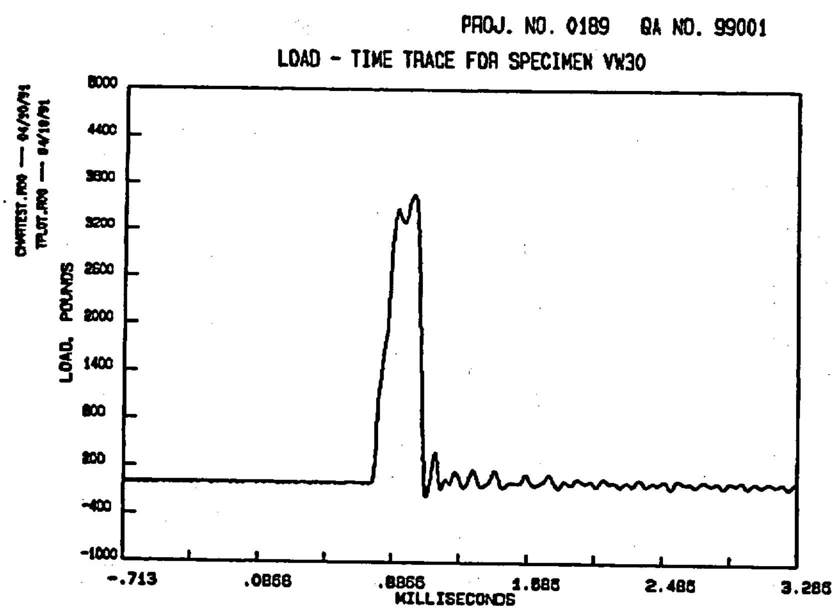

Figure B-24. Load-Time Trace for Charpy V-Notch Impact Specimen VW30

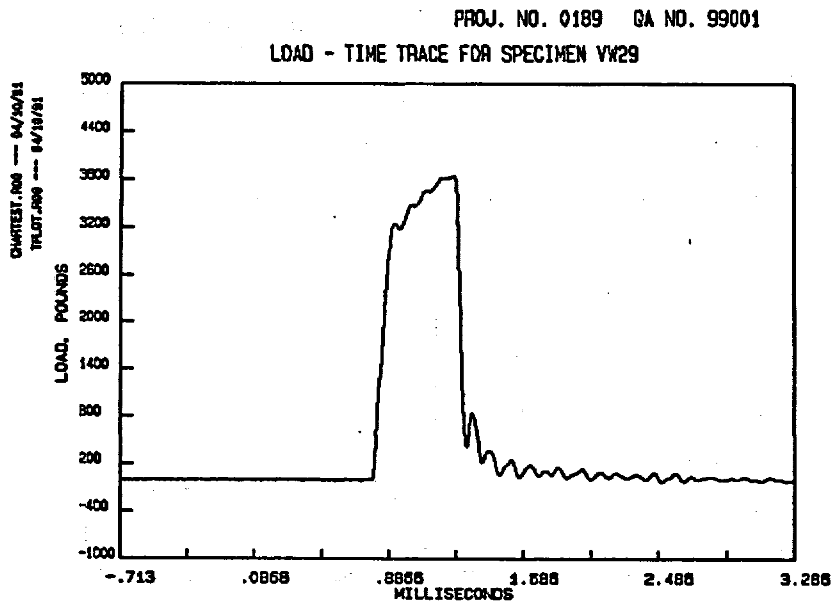

Figure B-25. Load-Time Trace for Charpy V-Notch Impact Specimen VW29

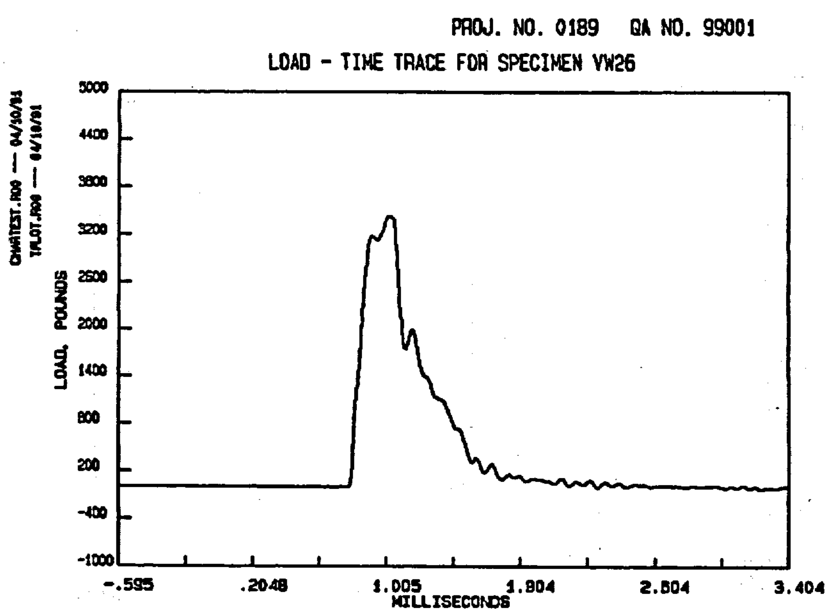

Figure B-26. Load-Time Trace for Charpy V-Notch Impact Specimen VW26

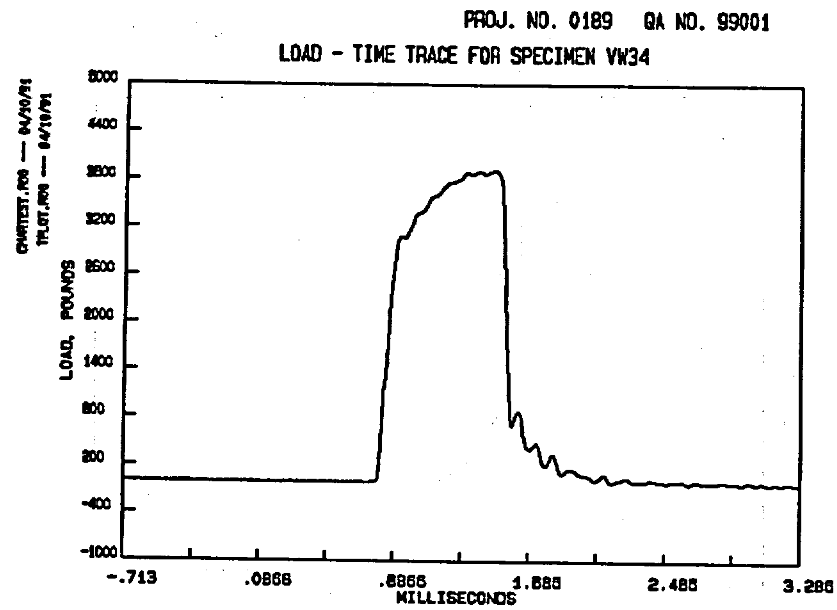

Figure B-27. Load-Time Trace for Charpy V-Notch Impact Specimen VW34

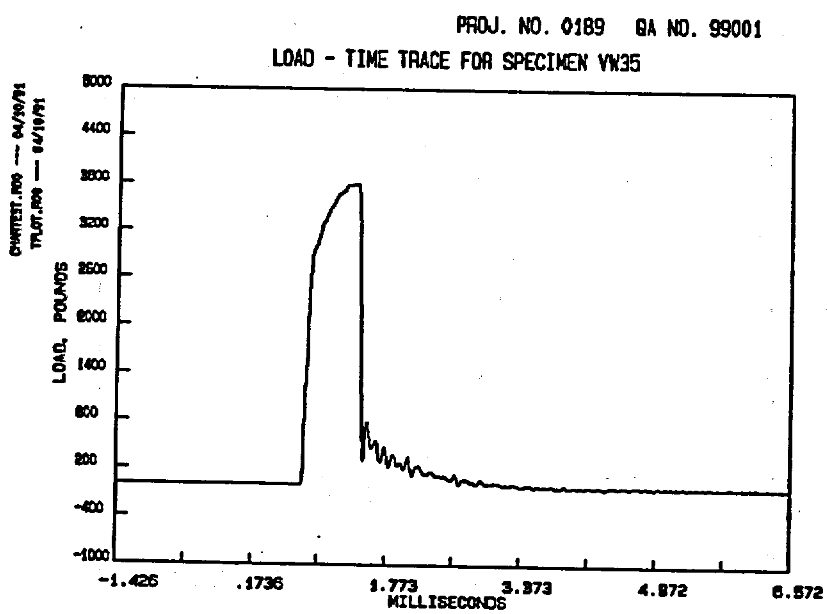

Figure B-28. Load-Time Trace for Charpy V-Notch Impact Specimen VW35

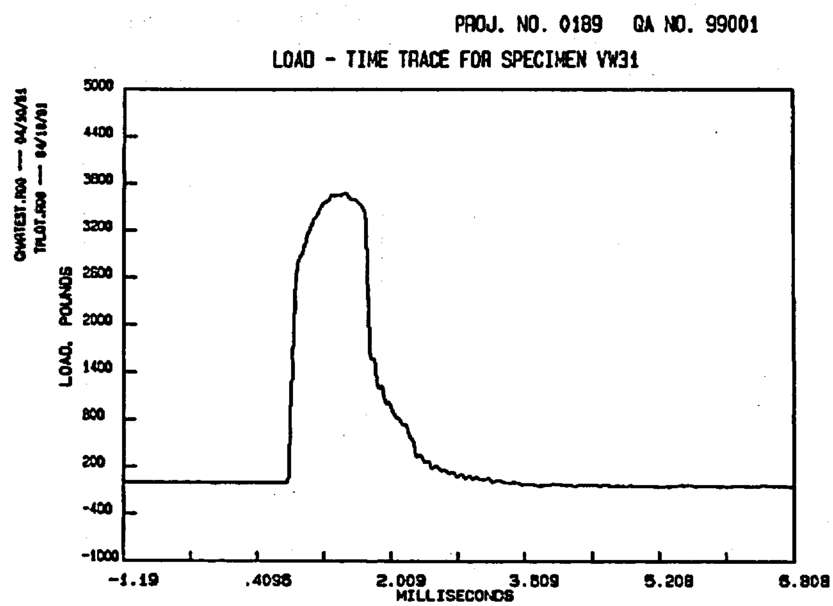

Figure B-29. Load-Time Trace for Charpy V-Notch Impact Specimen VW31

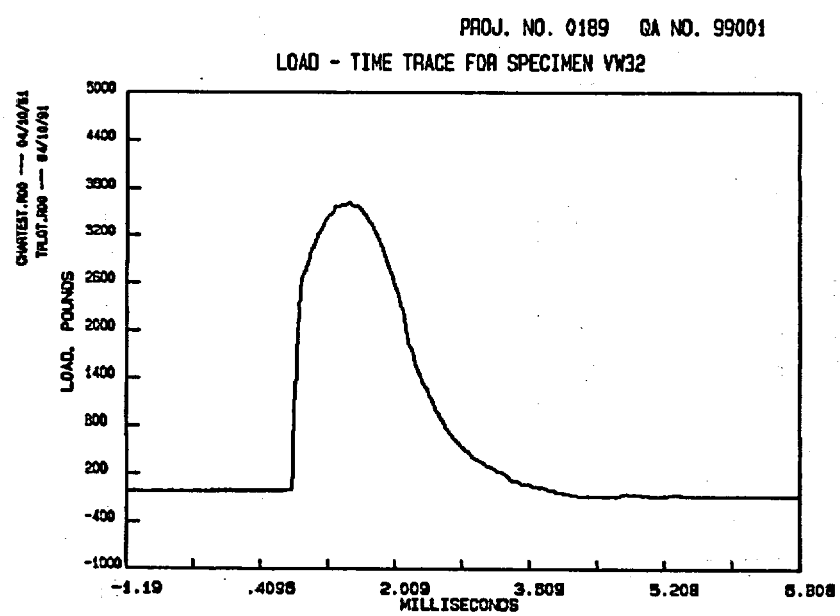

Figure B-30. Load-Time Trace for Charpy V-Notch Impact Specimen VW32

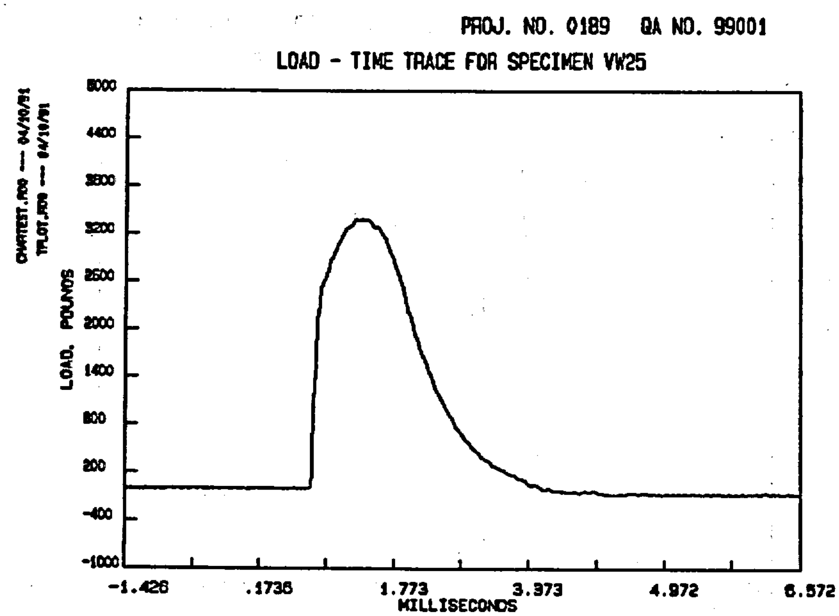

Figure B-31. Load-Time Trace for Charpy V-Notch Impact Specimen VW25

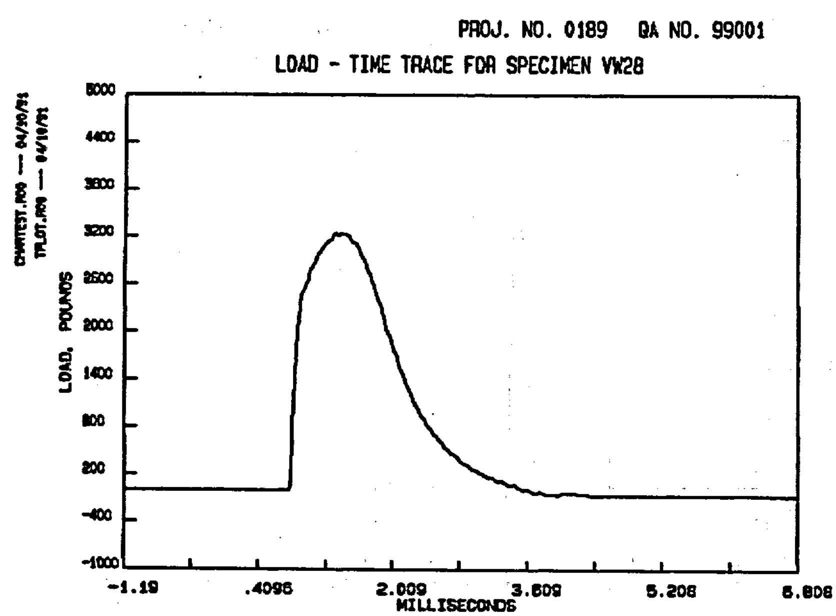

Figure B-32. Load-Time Trace for Charpy V-Notch Impact Specimen VW28

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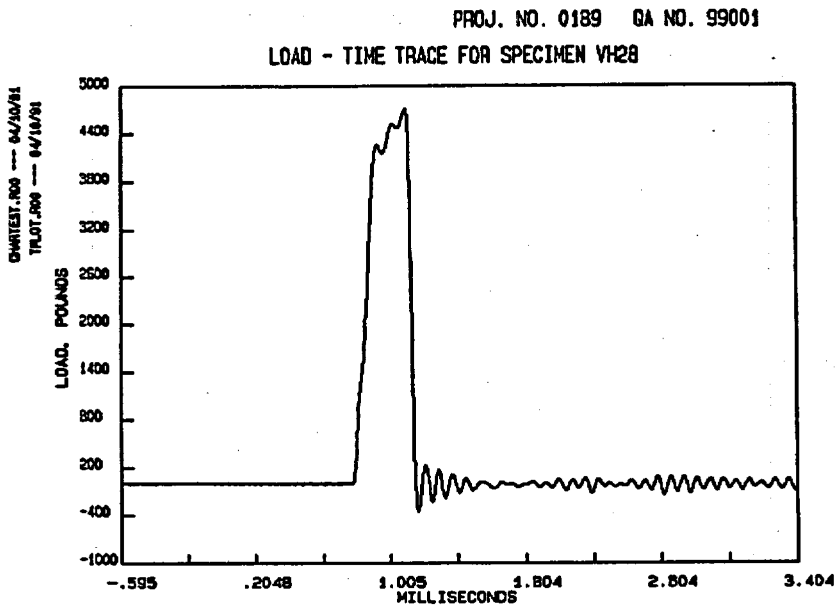

Figure B-33. Load-Time Trace for Charpy V-Notch Impact Specimen VH28

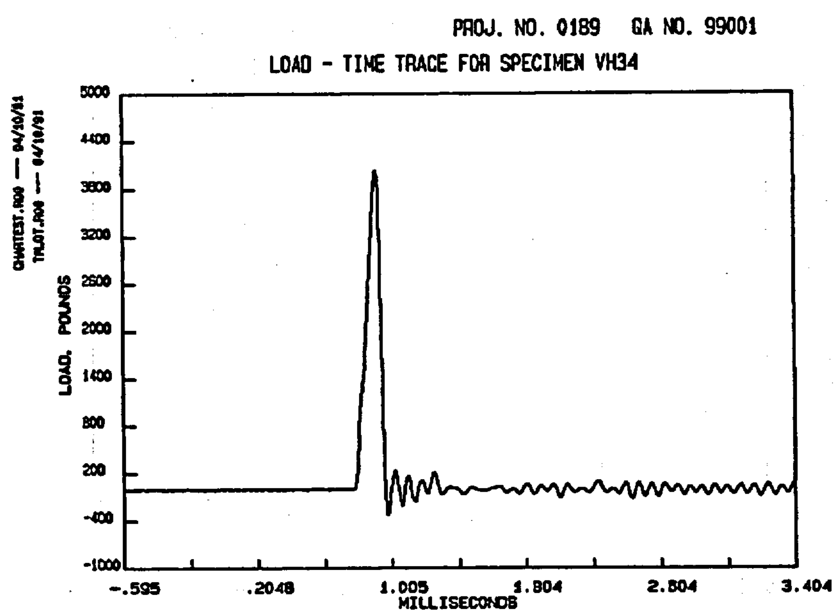

Figure B-34. Load-Time Trace for Charpy V-Notch Impact Specimen VH34

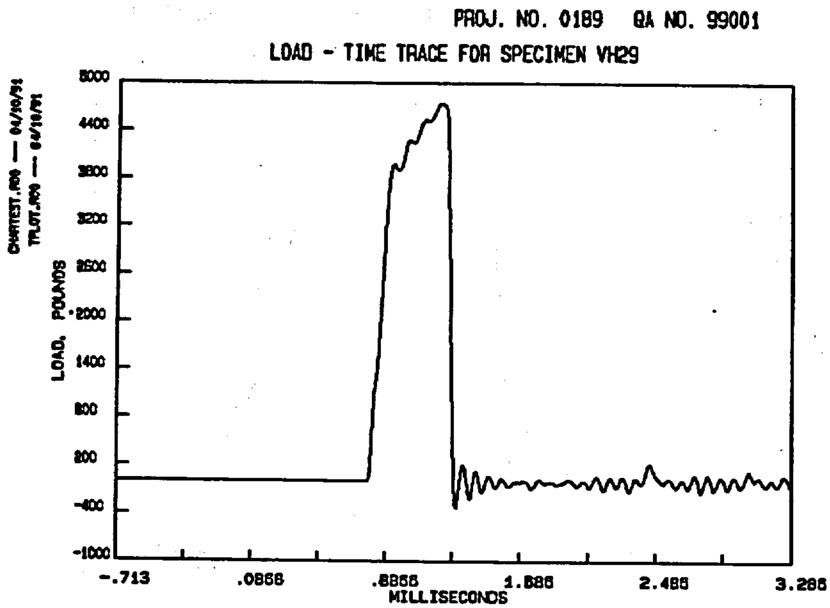

Figure B-35. Load-Time Trace for Charpy V-Notch Impact Specimen VH29

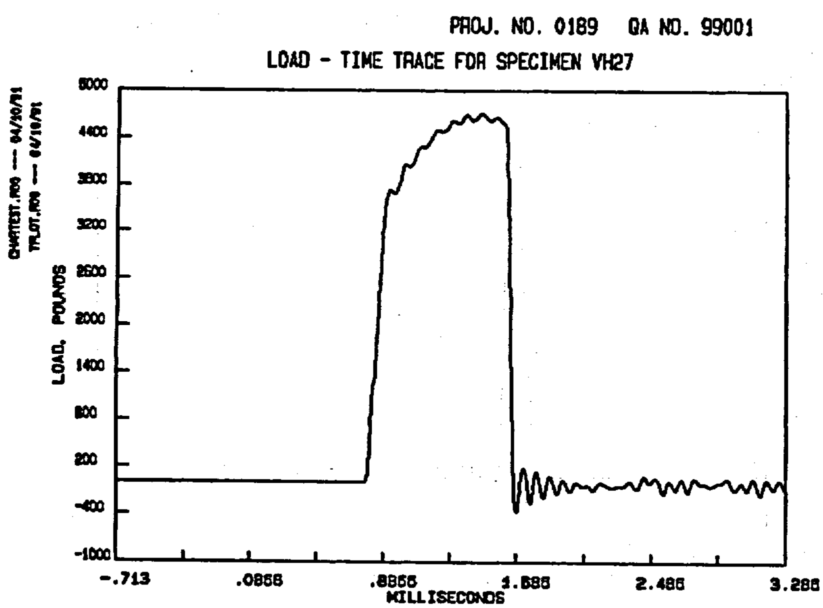

Figure B-36. Load-Time Trace for Charpy V-Notch Impact Specimen VH27

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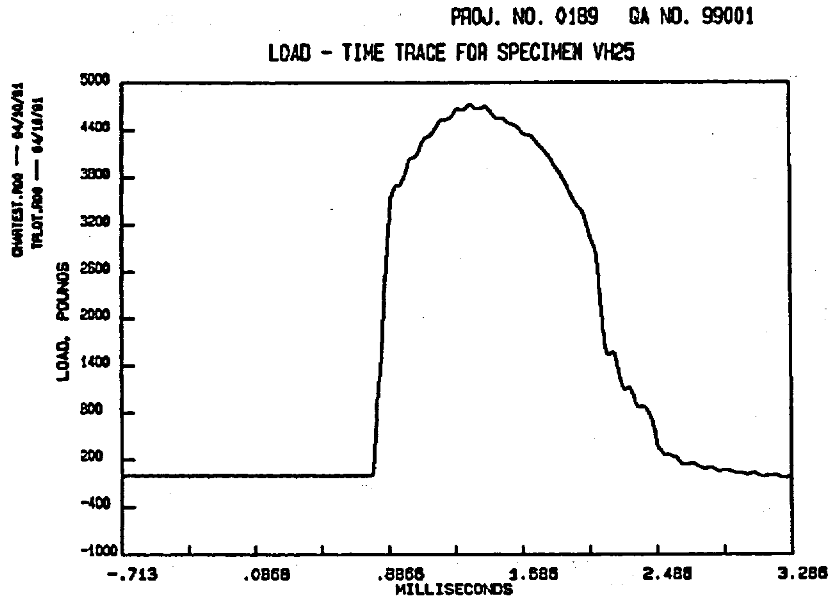

Figure B-37. Load-Time Trace for Charpy V-Notch Impact Specimen VH25

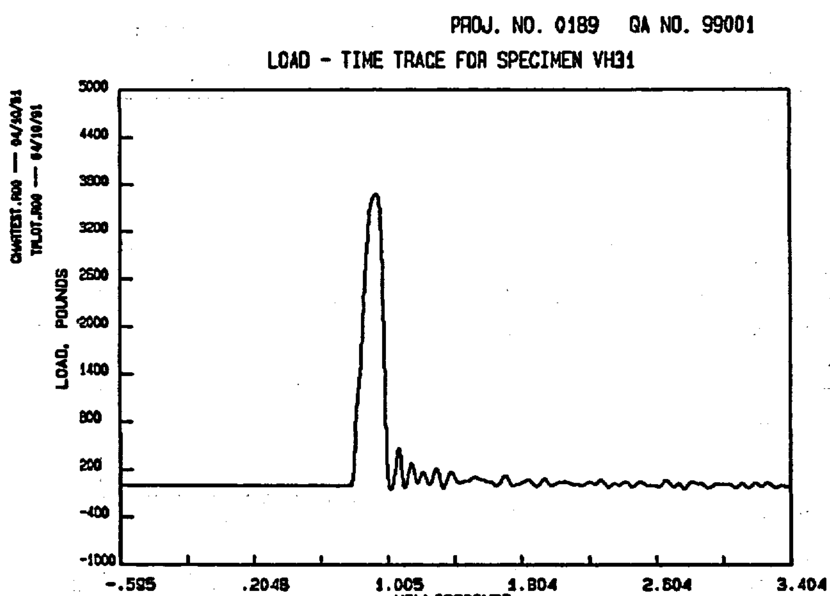

Figure B-38. Load-Time Trace for Charpy V-Notch Impact Specimen VH31

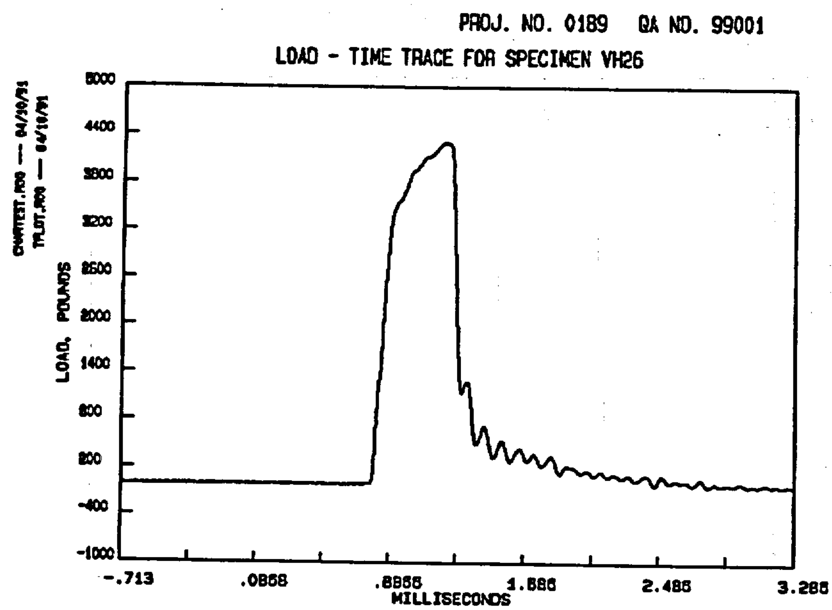

Figure B-39. Load-Time Trace for Charpy V-Notch Impact Specimen VH26

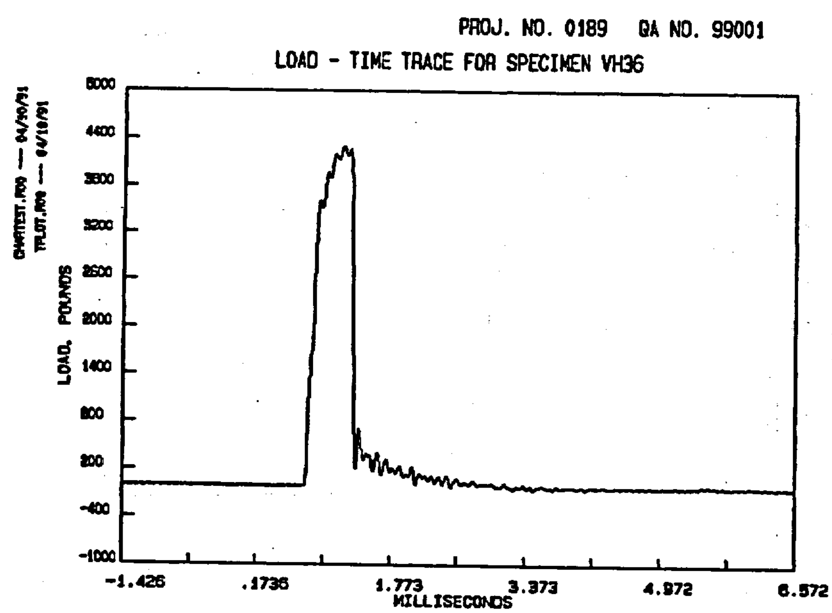

Figure B-40. Load-Time Trace for Charpy V-Notch Impact Specimen VH36

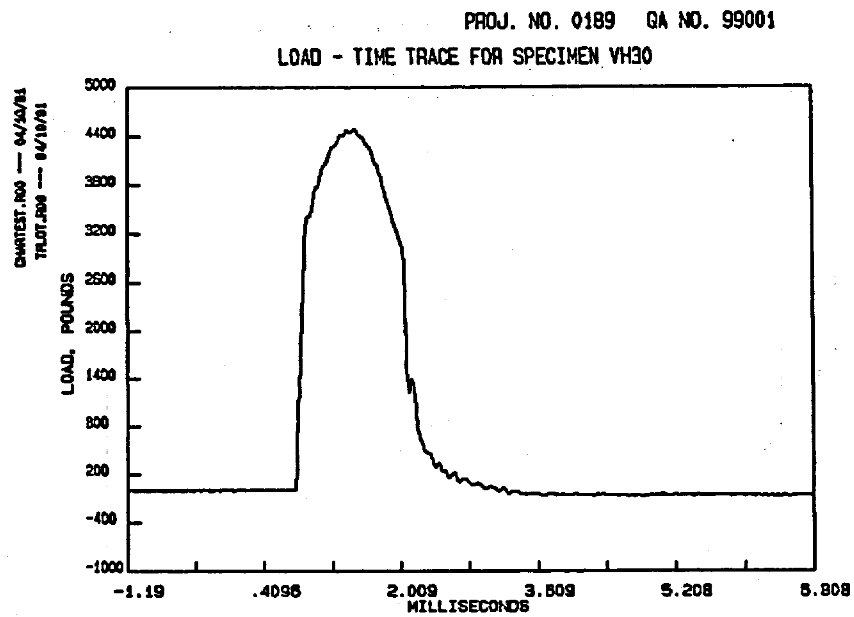

Figure B-41. Load-Time Trace for Charpy V-Notch Impact Specimen VH30

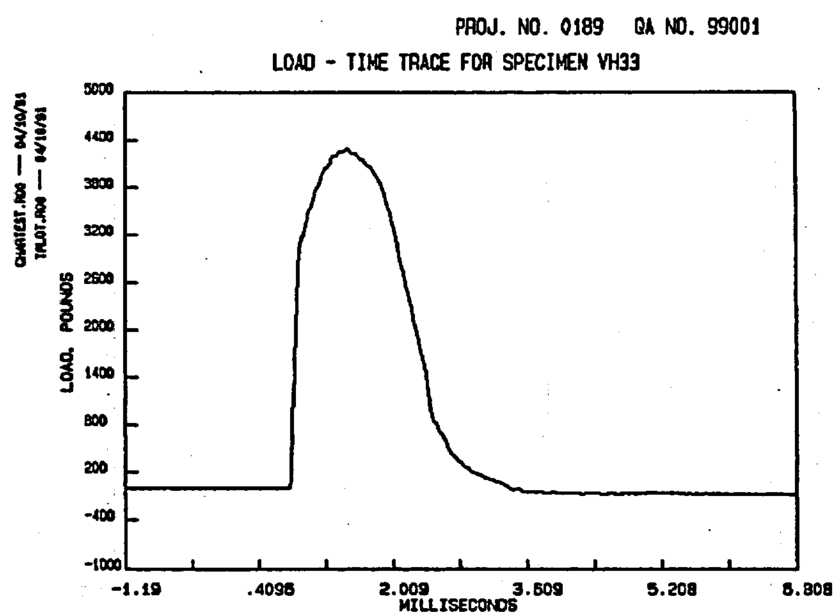

Figure B-42. Load-Time Trace for Charpy V-Notch Impact Specimen VH33

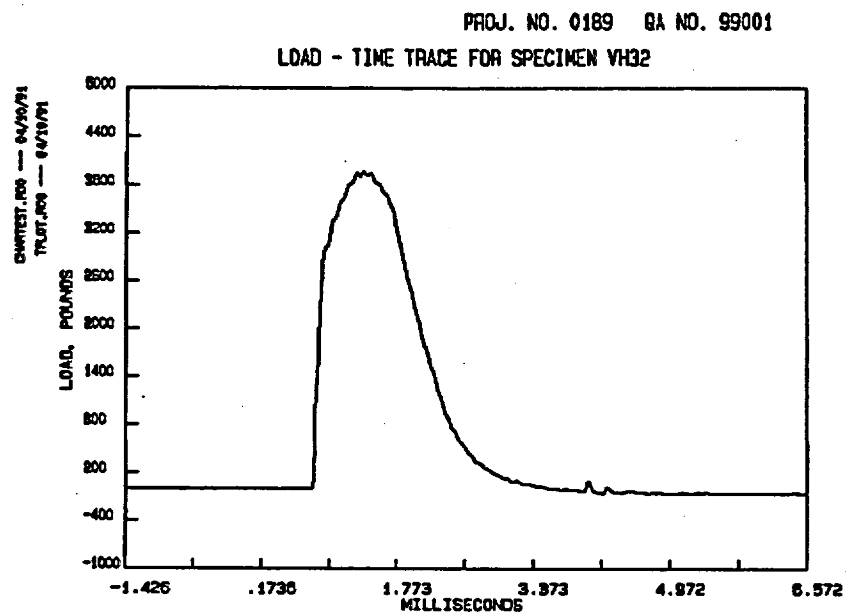

Figure B-43. Load-Time Trace for Charpy V-Notch Impact Specimen VH32

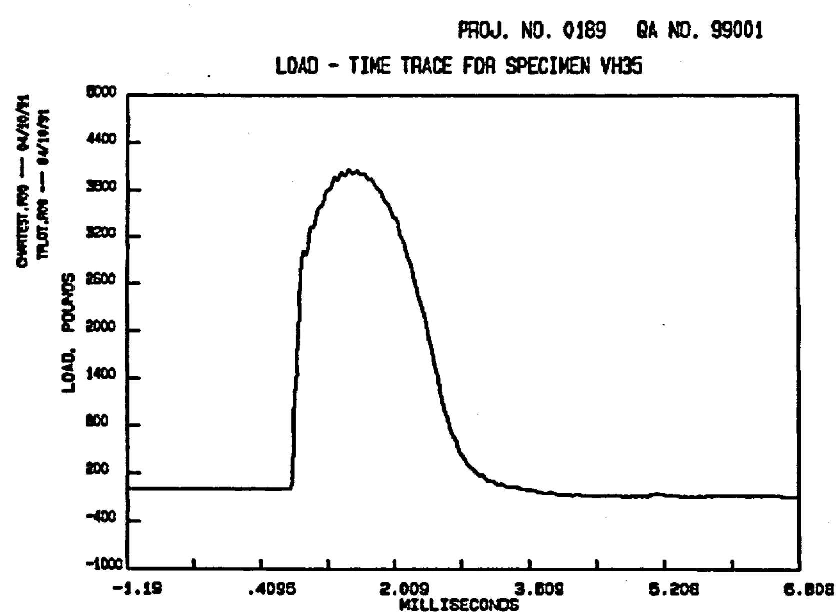

Figure B-44. Load-Time Trace for Charpy V-Notch Impact Specimen VH35

APPENDIX C

Unirradiated Tensile Data for the North Anna Unit No. 1 RVSP Materials

Table C-1. Tensile Properties of Unirradiated Base Metal Forging 03, Heat No. 990400/292332, Axial Orientation

Specimen No.	Test Temp. (F)	Strength, psi		Elongation, %		Reduction of Area, %
		Yield	Ultimate	Uniform	Total	
---	Room	70,050	92,404	11.4	18.8	60.7
---	Room	71,300	92,700	12.3	18.8	57.0
---	300	64,075	84,600	13.0	21.5	61.0
---	300	64,750	85,200	13.7	23.7	64.0
---	550	58,027	87,150	13.7	20.6	52.0
---	550	53,137	85,325	17.2	26.0	57.0

Table C-2. Tensile Properties of Unirradiated Weld Metal, Wire Heat 25531 / Flux Lot 1211

Specimen No.	Test Temp. (F)	Strength, psi		Elongation, %		Reduction of Area, %
		Yield	Ultimate	Uniform	Total	
---	Room	63,150	78,300	9.6	18.9	71.0
---	Room	65,200	80,400	9.8	19.5	71.0
---	300	64,300	77,875	8.6	19.5	67.0
---	300	59,675	73,625	10.8	22.5	68.9
---	550	60,175	76,850	9.8	20.0	63.0
---	550	61,650	80,500	8.9	18.0	57.0

APPENDIX D

**Unirradiated and Irradiated
Charpy V-Notch Impact Surveillance Data for the
North Anna Unit No. 1 RVSP Materials
Using Hyperbolic Tangent Curve-Fitting Method**

Table D-1. Unirradiated Surveillance Charpy V-Notch Impact Data for North Anna Unit No. 1, Base Metal Forging 03, Heat No. 990400/292332, Tangential Orientation

Specimen No.	Test Temp. (F)	Impact Energy (ft-lb)	Lateral Expansion (mils)	Shear Fracture (%)
---	-15	30	21	9
---	-15	9	5	0
---	-15	19	16	3
---	40	66	48	40
---	40	84	60	45
---	40	78	59	56
---	74	96.5	67	38
---	74	101	74	64
---	74	73.5	59	38
---	125	116	74	77
---	125	115	78	92
---	125	99	67.5	79
---	170	143	80	100
---	170	147	83	100
---	170	144	81	100
---	210	126.5	88	100
---	210	123	84.5	100
---	210	127	80	100

Table D-2. North Anna Unit No. 1 Capsule V Surveillance Charpy Impact Data for Base Metal Forging 03, Heat No. 990400/29232, Irradiated to 2.63 x 10^{18} n/cm^2 (E>1.0 MeV) Tangential Orientation

Specimen No.	Test Temp. (F)	Impact Energy (ft-lb)	Lateral Expansion (mils)	Shear Fracture (%)
VL15	0	7.0	4.0	0
VL13	40	36.0	26.5	0
VL9	60	51.0	40.0	5
VL16	73	32.0	23.0	25
VL12	100	50.0	39.0	43
VL10	120	76.5	57.5	56
VL11	196	113.0	78.0	89
VL14	280	122.0	86.0	100

Table D-3. North Anna Unit No. 1 Capsule U Surveillance Charpy Impact Data for Base Metal Forging 03, Heat No. 990400/292332, Irradiated to 8.72×10^{18} n/cm² (E>1.0 MeV) Tangential Orientation

Specimen No.	Test Temp. (F)	Impact Energy (ft-lb)	Lateral Expansion (mils)	Shear Fracture (%)
VL46	10	10.0	10.0	5
VL41	74	38.0	28.5	20
VL44	77	12.0	15.5	15
VL48	125	---[(a)]	---[(a)]	---[(a)]
VL45	125	17.0	23.0	20
VL43	175	75.0	62.0	70
VL42	250	110.0	82.0	100
VL47	350	109.0	81.0	100

(a) Machine malfunction.

Table D-4. Hyperbolic Tangent Curve Fit Coefficients for North Anna Unit No. 1, Base Metal Forging 03 Heat No. 990400/292332, Tangential Orientation

	Hyperbolic Tangent Curve Fit Coefficients		
	Absorbed Energy	Lateral Expansion	Percent Shear Fracture
Unirradiated	A: 67.5 B: 65.3 C: 73.9 T0: 42.9	A: 40.4 B: 39.4 C: 52.5 T0: 24.6	A: 50.0 B: 50.0 C: 76.0 T0: 64.2
Capsule V	A: 65.1 B: 62.9 C: 99.8 T0: 108.8	A: 44.7 B: 43.7 C: 98.5 T0: 100.2	A: 50.0 B: 50.0 C: 53.4 T0: 111.8
Capsule U	A: 59.9 B: 57.7 C: 83.7 T0: 159.1	A: 44.1 B: 43.1 C: 99.1 T0: 143.1	A: 50.0 B: 50.0 C: 68.3 T0: 151.6

Table D-5. Unirradiated Surveillance Charpy V-Notch Impact Data for North Anna Unit No. 1, Base Metal Forging 03, Heat No. 990400/292332, Axial Orientation

Specimen No.	Test Temp. (F)	Impact Energy (ft-lb)	Lateral Expansion (mils)	Shear Fracture (%)
---	-15	16	7	10
---	-15	17	10	10
---	-15	15	10	10
---	45	31.5	21	35
---	45	30	20	35
---	45	25	21	30
---	75	47.5	37	30
---	75	43.5	37	23
---	75	47.5	33	27
---	105	62.5	50	55
---	105	56.5	50	55
---	105	59	50	55
---	150	80.5	66	90
---	150	84.5	72	100
---	150	67.5	62	78
---	210	85	69	100
---	210	82.5	68	100
---	210	86.5	73	100

Table D-6. North Anna Unit No. 1 Capsule V Surveillance Charpy Impact Data for Base Metal Forging 03, Heat No. 990400/292332, Irradiated to 2.63 x 10^{18} n/cm^2 (E>1.0 MeV) Axial Orientation

Specimen No.	Test Temp. (F)	Impact Energy (ft-lb)	Lateral Expansion (mils)	Shear Fracture (%)
VT19	0	12.5	7.0	0
VT14	40	15.5	11.5	0
VT16	60	29.0	23.5	14
VT13	73	39.0	29.0	0
VT22	100	40.0	32.0	34
VT15	120	41.0	35.0	42
VT24	130	51.0	42.0	38
VT18	140	59.0	48.0	98
VT23	160	60.0	50.0	98
VT21	195	79.0	66.0	100
VT20	240	68.5	60.0	100
VT17	280	77.0	68.0	100

Table D-7. North Anna Unit No. 1 Capsule U Surveillance Charpy Impact Data for Base Metal Forging 03, Heat No. 990400/292332, Irradiated to 8.72 x 10^{18} n/cm^2 (E>1.0 MeV) Axial Orientation

Specimen No.	Test Temp. (F)	Impact Energy (ft-lb)	Lateral Expansion (mils)	Shear Fracture (%)
VT70	0	2.0	2.5	2
VT67	74	20.0	20.0	10
VT63	100	28.0	29.0	15
VT69	100	26.0	30.0	15
VT64	125	42.0	39.0	25
VT68	125	33.0	35.0	20
VT71	150	47.0	44.0	35
VT66	200	45.0	42.0	45
VT62	225	76.0	56.5	80
VT61	275	98.0	70.0	100
VT65	350	90.0	75.0	100
VT72	400	90.0	66.0	100

Table D-8. Hyperbolic Tangent Curve Fit Coefficients for North Anna Unit No. 1 Base Metal Forging 03, Heat No. 990400/292332, Axial Orientation

	Hyperbolic Tangent Curve Fit Coefficients		
	Absorbed Energy	Lateral Expansion	Percent Shear Fracture
Unirradiated	A: 46.6 B: 44.4 C: 93.4 T0: 76.4	A: 37.3 B: 36.3 C: 73.6 T0: 76.9	A: 50.0 B: 50.0 C: 74.8 T0: 93.0
Capsule V	A: 40.4 B: 38.2 C: 100.0 T0: 97.3	A: 34.9 B: 33.9 C: 98.8 T0: 106.0	A: 50.0 B: 50.0 C: 32.4 T0: 121.8
Capsule U	A: 48.9 B: 46.7 C: 116.8 T0: 162.5	A: 36.3 B: 35.3 C: 128.8 T0: 134.2	A: 50.0 B: 50.0 C: 86.7 T0: 181.2

Table D-9. Unirradiated Surveillance Charpy V-Notch Impact Data for North Anna Unit No. 1, Weld Metal, Wire Heat 25531 / Flux Lot 1211

Specimen No.	Test Temp. (F)	Impact Energy (ft-lb)	Lateral Expansion (mils)	Shear Fracture (%)
---	-110	9	9	0
---	-110	5	9	0
---	-110	13	7	0
---	0	74.5	61	48
---	0	58	52	54
---	0	60	70	43
---	48	37.5	39.5	31
---	48	40	45	51
---	48	42	49.5	40
---	75	83	68	79
---	75	81	71.5	81
---	75	49.5	50	65
---	175	106	89	100
---	175	106	88	100
---	175	94	76	96
---	250	87	79	100
---	250	90.5	81	100
---	250	84.5	77	100

Table D-10. North Anna Unit No. 1 Capsule V Surveillance Charpy Impact Data for Weld Metal (Wire Heat 25531 / Flux Lot 1211), Irradiated to 2.63 x 10^{18} n/cm^2 (E>1.0 MeV)

Specimen No.	Test Temp. (F)	Impact Energy (ft-lb)	Lateral Expansion (mils)	Shear Fracture (%)
VW20	-40	4.5	4.5	0
VW22	0	25.0	25.5	50
VW24	40	26.0	29.0	46
VW17	60	33.0	31.5	28
VW16	73	39.0	31.0	34
VW23	90	47.0	49.0	89
VW21	100	60.0	56.0	25
VW14	120	73.0	62.0	72
VW15	160	62.5	52.5	96
VW18	198	92.5	73.0	100
VW13	240	100.0	80.0	100
VW19	280	87.0	77.0	100

Table D-11. North Anna Unit No. 1 Capsule U Surveillance Charpy Impact Data for Weld Metal (Wire Heat 25531 / Flux Lot 1211), Irradiated to 8.72 x 10^{18} n/cm^2 (E>1.0 MeV)

Specimen No.	Test Temp. (F)	Impact Energy (ft-lb)	Lateral Expansion (mils)	Shear Fracture (%)
VW71	-60	---[a]	---[a]	---[a]
VW63	-25	38.0	33.5	30
VW61	25	25.0	28.0	20
VW71	25	34.0	40.5	25
VW64	40	50.0	45.5	45
VW68	60	53.0	48.0	45
VW69	74	30.0	30.0	45
VW65	125	51.0	52.5	50
VW70	200	71.0	70.0	85
VW66	275	95.0	85.0	100
VW62	350	87.0	77.0	100
VW67	400	94.0	84.5	100

(a) Machine malfunction.

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Table D-12. Hyperbolic Tangent Curve Fit Coefficients for North Anna Unit No. 1 Weld Metal (Wire Heat 25531 / Flux Lot 1211)

	Hyperbolic Tangent Curve Fit Coefficients		
	Absorbed Energy	Lateral Expansion	Percent Shear Fracture
Unirradiated	A: 49.9 B: 47.7 C: 136.7 T0: 16.9	A: 41.5 B: 40.5 C: 126.4 T0: -5.8	A: 50.0 B: 50.0 C: 106.7 T0: 31.9
Capsule V	A: 49.1 B: 46.9 C: 101.2 T0: 88.2	A: 40.5 B: 39.5 C: 115.2 T0: 75.9	A: 50.0 B: 50.0 C: 112.3 T0: 72.3
Capsule U	A: 54.6 B: 52.4 C: 258.9 T0: 117.2	A: 46.3 B: 45.3 C: 245.7 T0: 78.8	A: 50.0 B: 50.0 C: 153.9 T0: 88.0

Table D-13. Unirradiated Surveillance Charpy V-Notch Impact Data for North Anna Unit No. 1, Heat-Affected-Zone Material

Specimen No.	Test Temp. (°F)	Impact Energy (ft-lb)	Lateral Expansion (mils)	Shear Fracture (%)
---	-125	21	13	9
---	-125	4.5	2.5	5
---	-125	13	6	5
---	-80	25.5	13	3
---	-80	76.5	47	29
---	-80	53	21	18
---	-25	22.5	13	18
---	-25	47	25	23
---	-25	31.5	15	13
---	40	104	65	81
---	40	62	43.5	45
---	40	78.5	46	42
---	100	156.5	78	100
---	100	127.5	81	79
---	100	125.5	73	90
---	170	152.5	72	100
---	170	166	82	100
---	170	109	74	100

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Table D-14. North Anna Unit No. 1 Capsule V Surveillance Charpy Impact Data for Heat-Affected-Zone Material, Irradiated to 2.63 x 10^{18} n/cm^2 (E > 1.0 MeV)

Specimen No.	Test Temp. (°F)	Impact Energy (ft-lb)	Lateral Expansion (mils)	Shear Fracture (%)
VH18	-40	66.0	48.0	3
VH13	0	19.5	8.0	10
VH16	40	22.5	15.0	60
VH19	50	16.5	15.0	40
VH20	60	49.5	30.5	89
VH23	73	89.0	66.0	62
VH21	100	81.0	55.0	71
VH15	120	67.5	42.0	55
VH24	160	93.0	67.0	99
VH22	198	105.0	71.0	100
VH14	240	113.0	70.0	100
VH17	280	102.0	74.0	100

Table D-15. North Anna Unit No. 1 Capsule U Surveillance Charpy Impact Data for Heat-Affected-Zone Material, Irradiated to 8.72 x 10^{18} n/cm^2 (E>1.0 MeV)

Specimen No.	Test Temp. (°F)	Impact Energy (ft-lb)	Lateral Expansion (mils)	Shear Fracture (%)
VH68	-100	25.0	21.5	10
VH67	-60	55.0	44.5	45
VH65	0	14.0	12.5	20
VH62	0	42.0	35.0	35
VH63	40	70.0	52.0	65
VH70	40	57.0	46.0	55
VH69	74	55.0	35.0	50
VH71	74	18.0	16.0	15
VH61	125	54.0	46.0	65
VH66	200	71.0	66.0	95
VH64	275	120.0	83.0	100
VH72	400	125.0	71.0	100

Table D-16. Hyperbolic Tangent Curve Fit Coefficients for North Anna Unit No. 1 Heat-Affected-Zone Material

Weld Metal	Hyperbolic Tangent Curve Fit Coefficients		
	Absorbed Energy	Lateral Expansion	Percent Shear Fracture
Unirradiated	A: 87.2 B: 85.0 C: 140.3 T0: 38.6	A: 44.1 B: 43.1 C: 119.1 T0: 16.8	A: 50.0 B: 50.0 C: 82.6 T0: 26.3
Capsule V	A: 64.4 B: 62.2 C: 172.7 T0: 88.2	A: 44.4 B: 43.4 C: 183.5 T0: 90.9	A: 50.0 B: 50.0 C: 87.8 T0: 47.8
Capsule U	A: 113.6 B: 111.4 C: 419.6 T0: 336.1	A: 47.8 B: 46.8 C: 340.1 T0: 116.2	A: 50.0 B: 50.0 C: 189.3 T0: 60.9

Figure D-1. Unirradiated Surveillance Charpy V-Notch Impact Data for North Anna Unit No. 1, Base Metal Forging 03, Heat No. 990400/292332, Tangential Orientation - Refitted Using Hyperbolic Tangent Curve-Fitting Method -

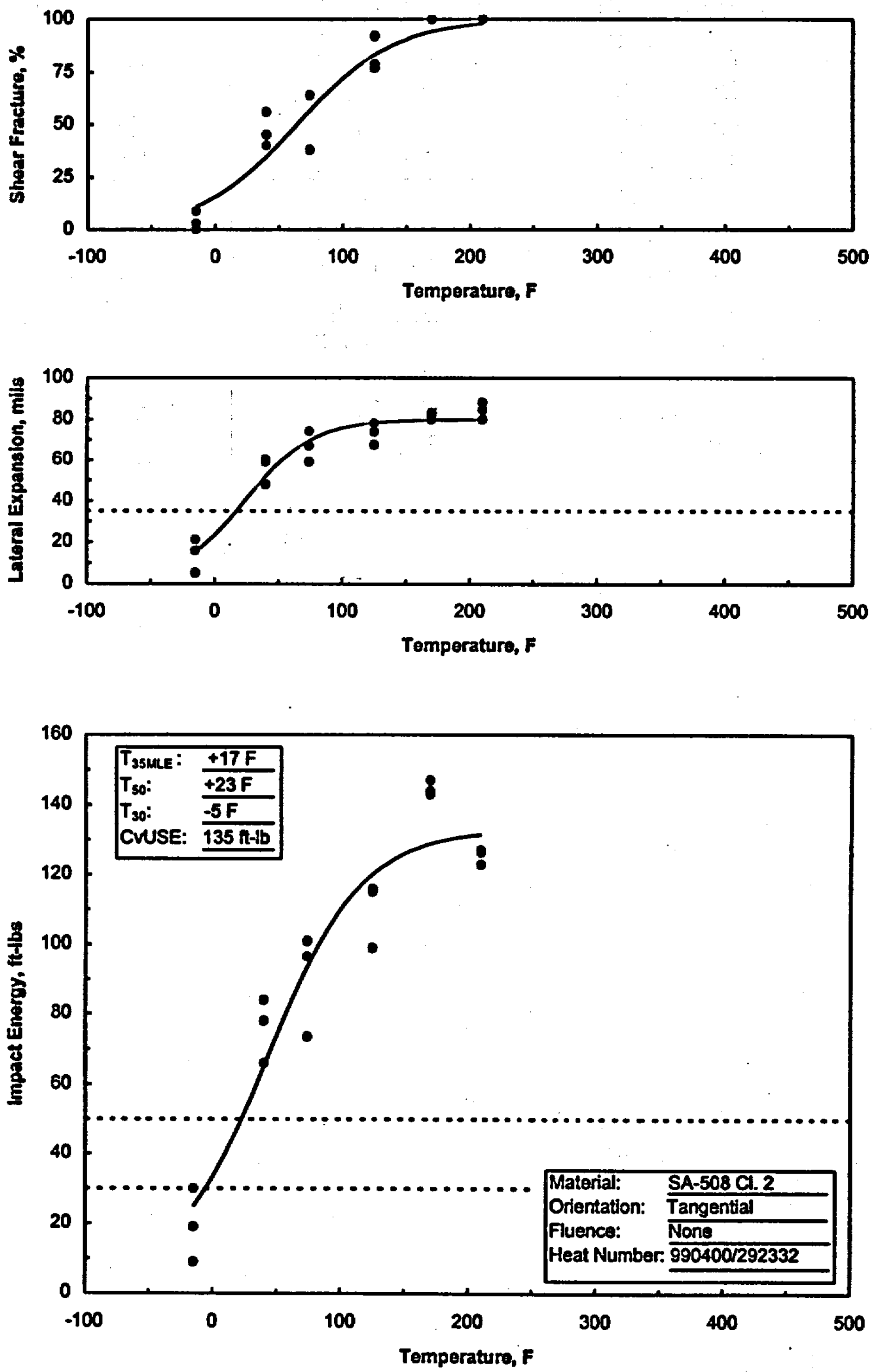

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Figure D-2. North Anna Unit No. 1 Capsule V Surveillance Charpy Impact Data for Base Metal Forging 03, Heat No. 990400/292332, Tangential Orientation
- Refitted Using Hyperbolic Tangent Curve-Fitting Method -

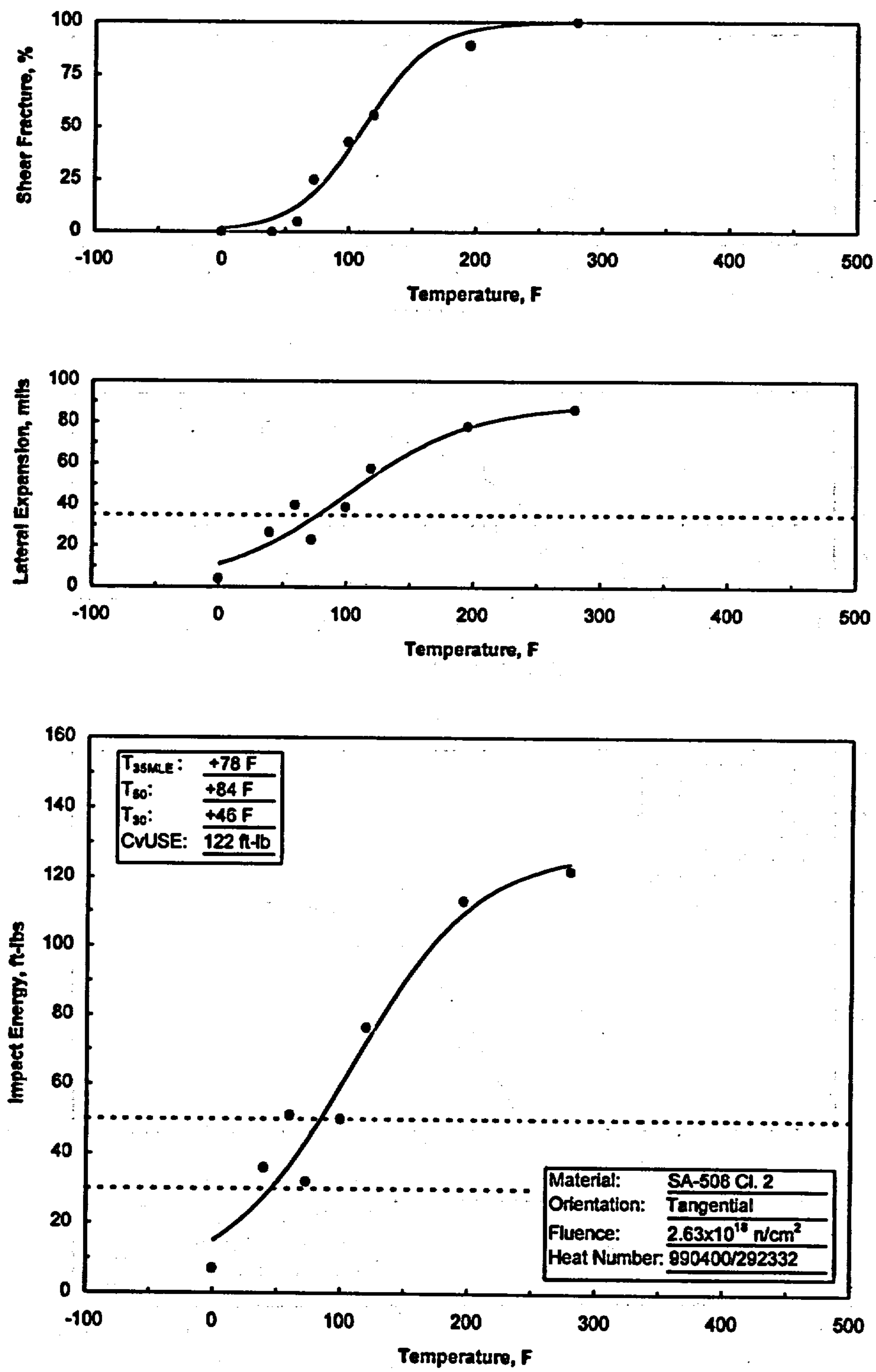

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Figure D-3. North Anna Unit No. 1 Capsule U Surveillance Charpy Impact Data for Base Metal Forging 03, Heat No. 990400/292332, Tangential Orientation
- Refitted Using Hyperbolic Tangent Curve-Fitting Method -

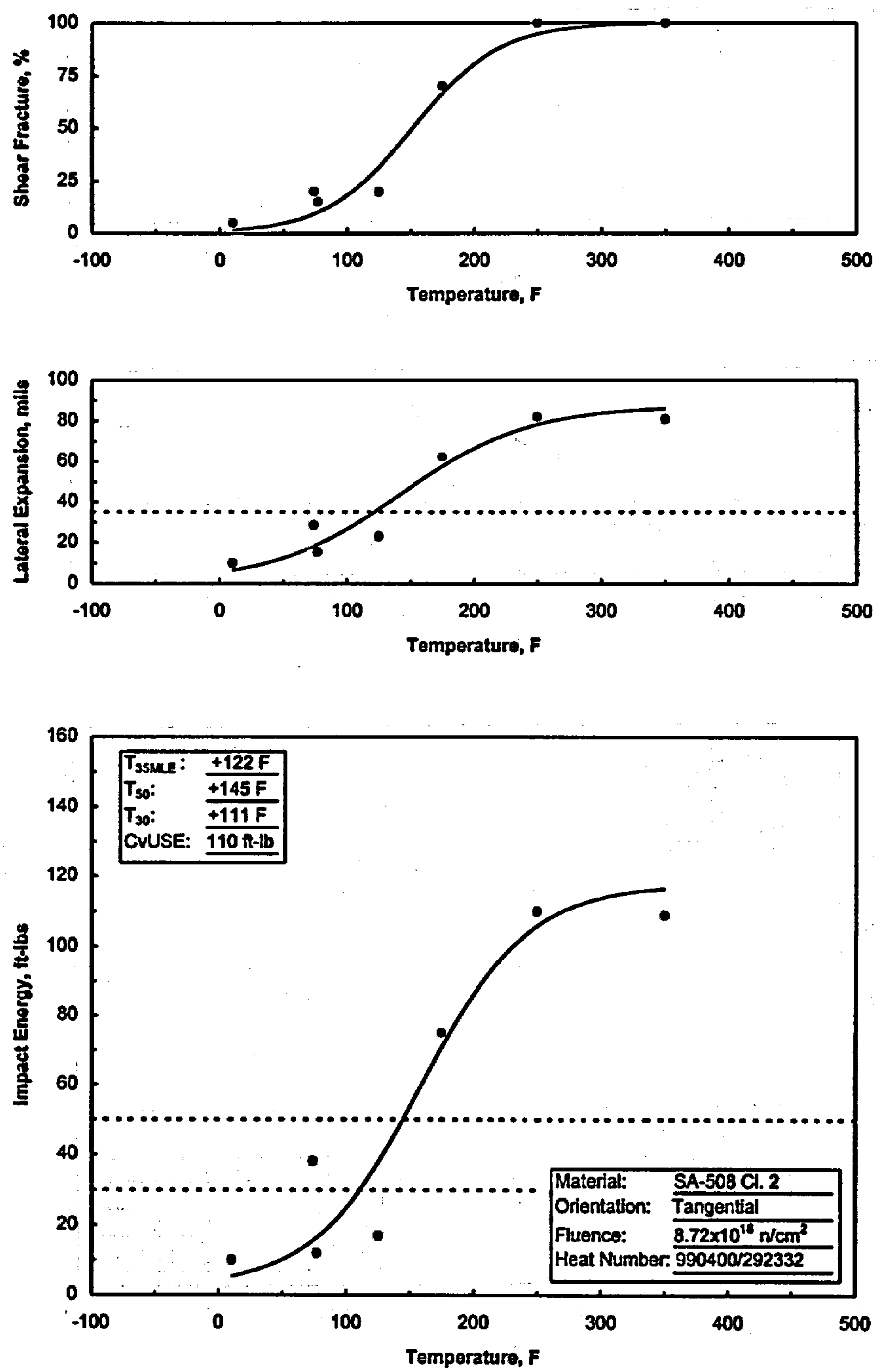

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Figure D-4. Unirradiated Surveillance Charpy V-Notch Impact Data for North Anna Unit No. 1, Base Metal Forging 03, Heat No. 990400/292332, Axial Orientation - Refitted Using Hyperbolic Tangent Curve-Fitting Method -

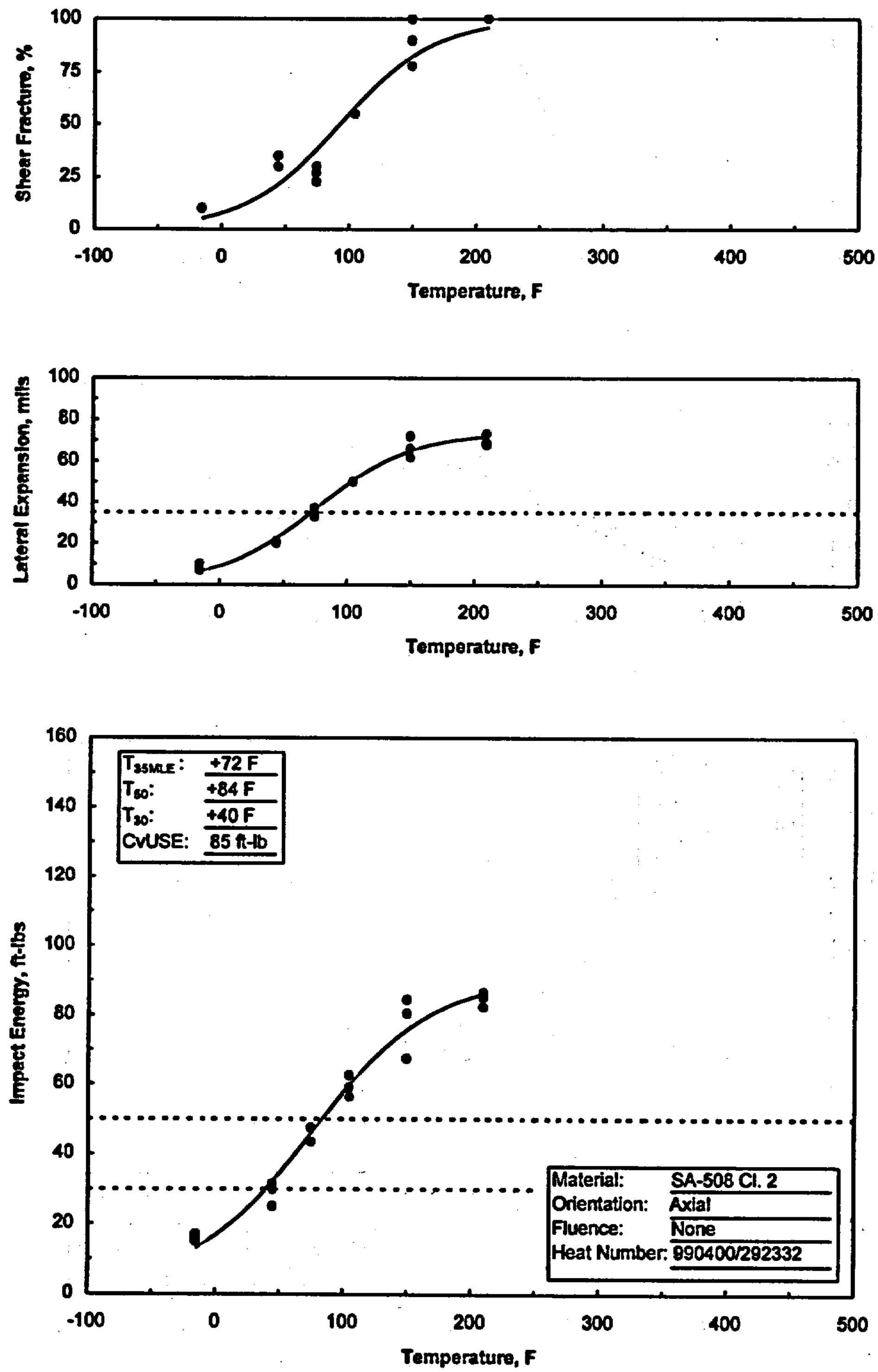
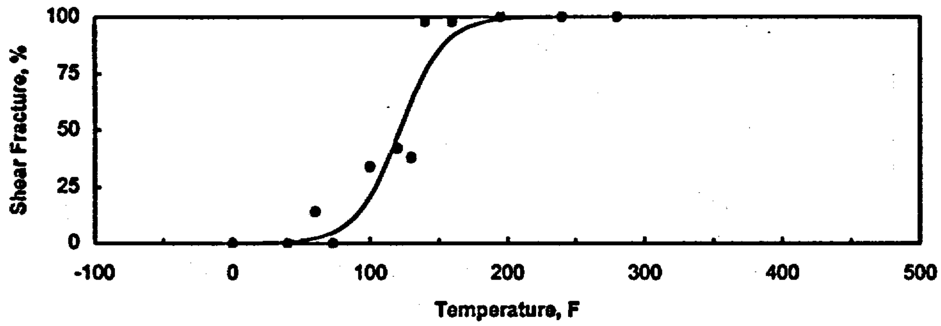
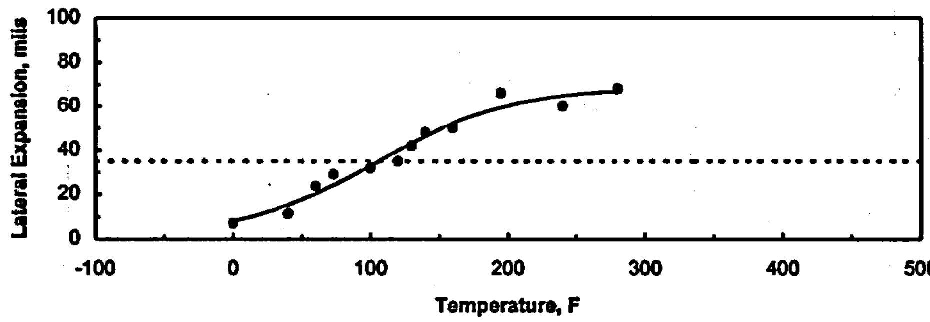


Figure D-5. North Anna Unit No. 1 Capsule V Surveillance Charpy Impact Data for Base Metal Forging 03, Heat No. 990400/292332, Axial Orientation - Refitted Using Hyperbolic Tangent Curve-Fitting Method -





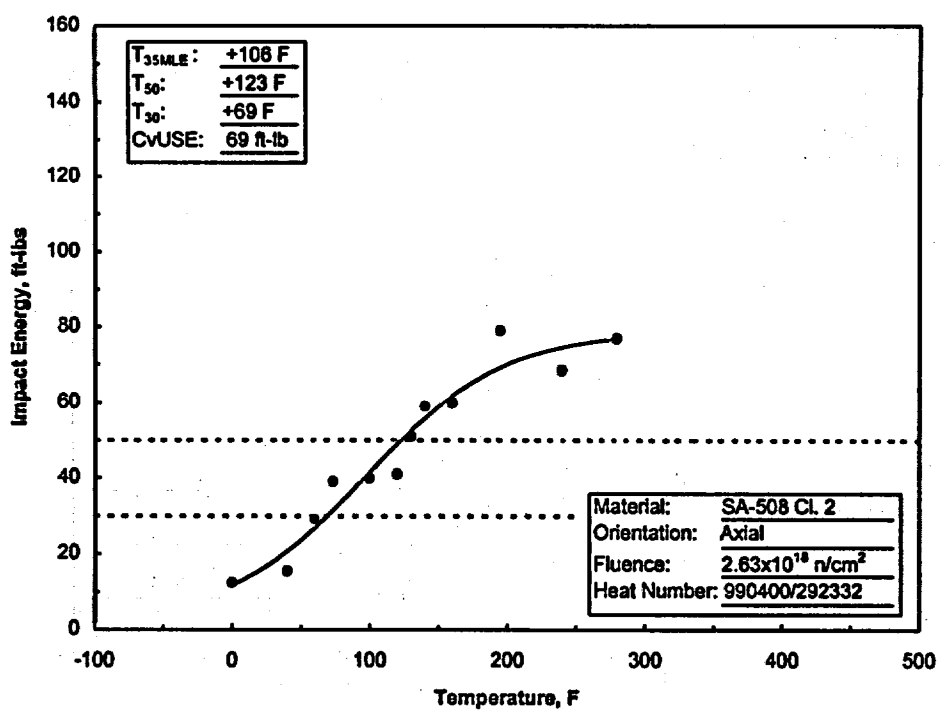


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Figure D-6. North Anna Unit No. 1 Capsule U Surveillance Charpy Impact Data for Base Metal Forging 03, Heat No. 990400/292332, Axial Orientation
- Refitted Using Hyperbolic Tangent Curve-Fitting Method -

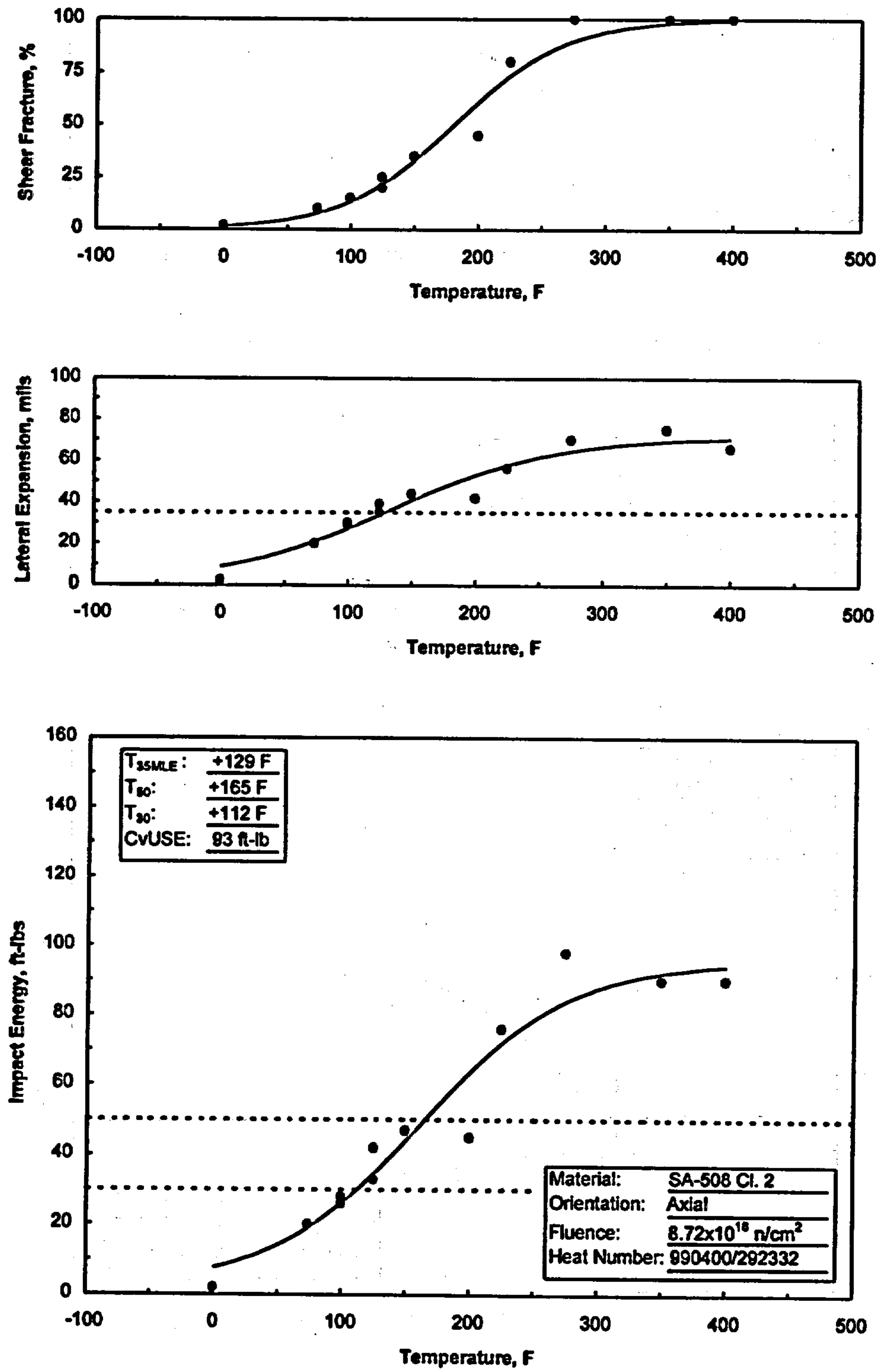

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Figure D-7. Unirradiated Surveillance Charpy V-Notch Impact Data for North Anna Unit No. 1, Weld Metal (Wire Heat 25531 / Flux Lot 1211) - Refitted Using Hyperbolic Tangent Curve-Fitting Method -

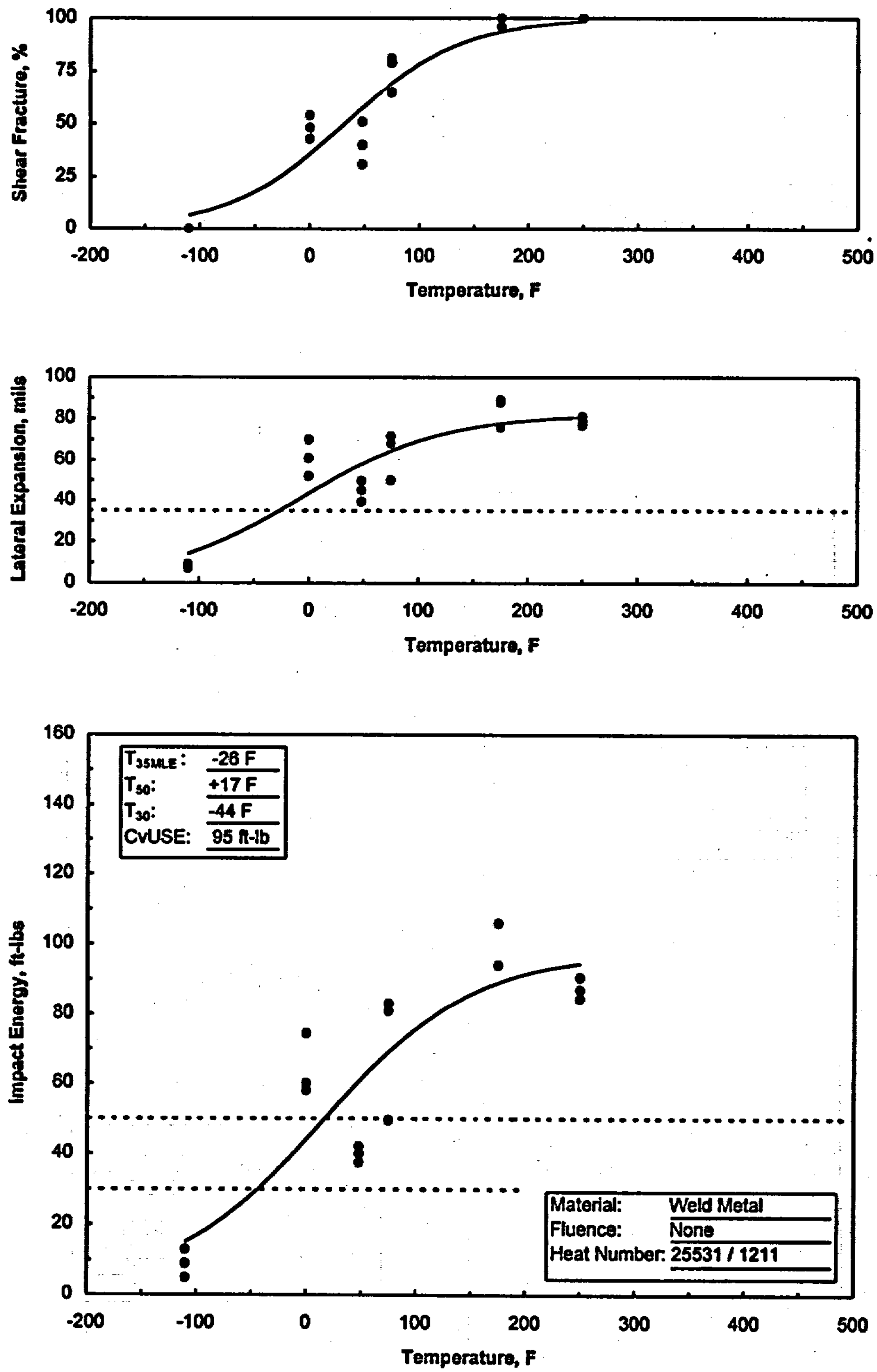

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Figure D-8. North Anna Unit No. 1 Capsule V Surveillance Charpy Impact Data for Weld Metal (Wire Heat 25531 / Flux Lot 1211) - Refitted Using Hyperbolic Tangent Curve-Fitting Method -

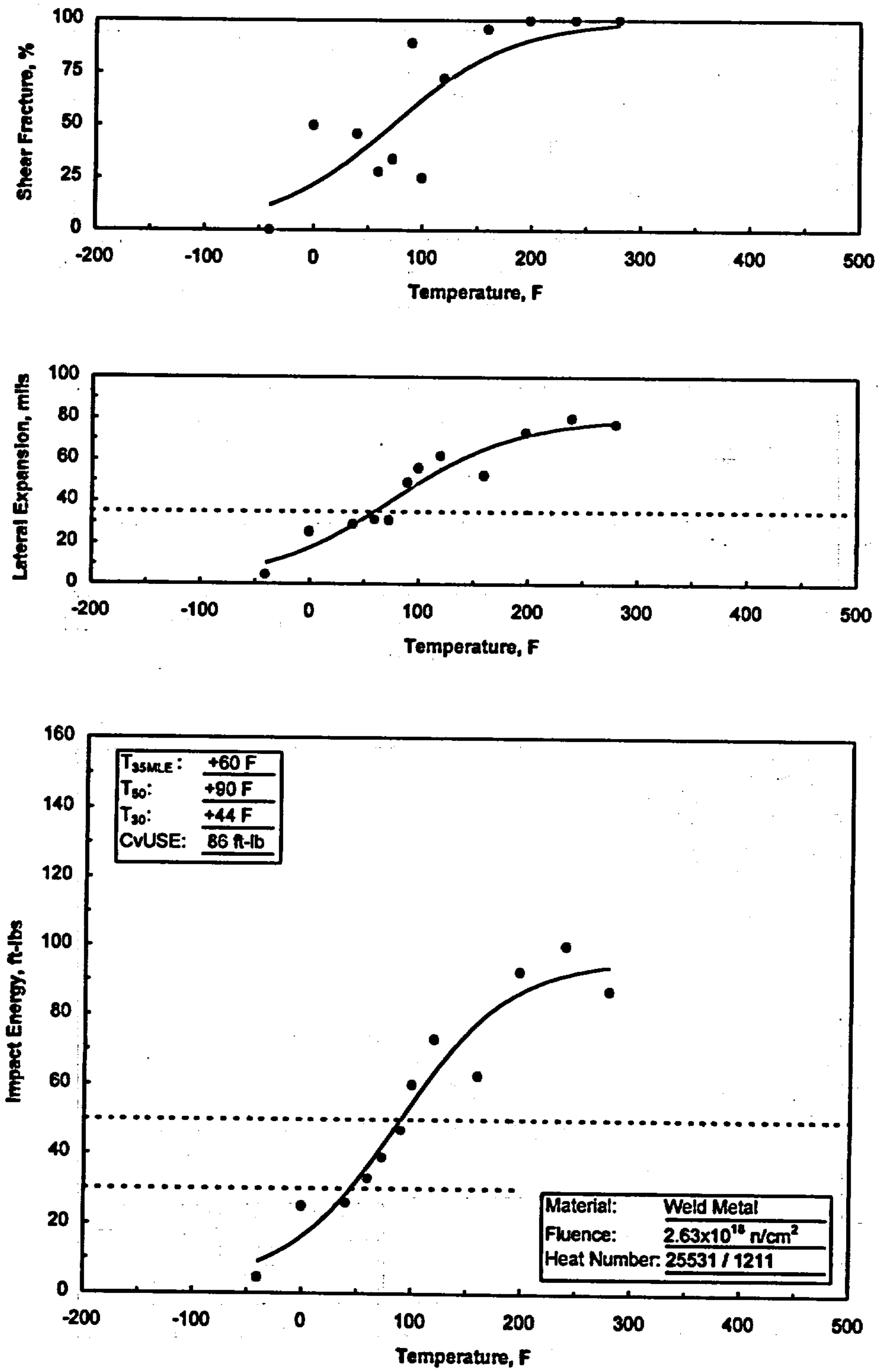
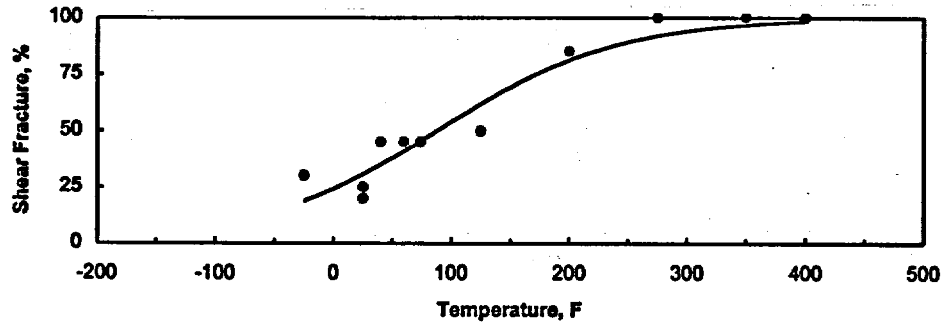
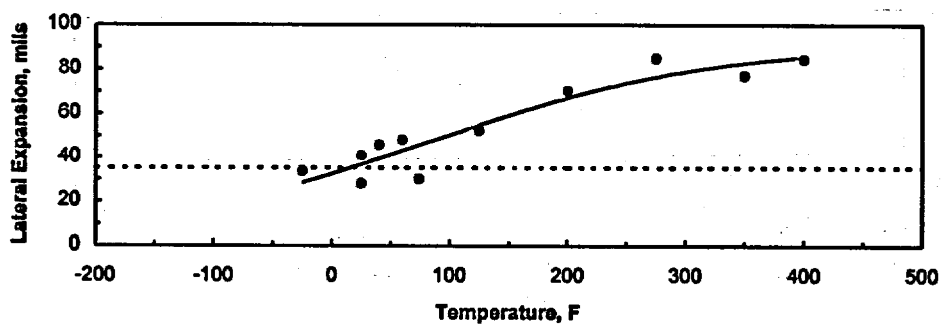


Figure D-9. North Anna Unit No. 1 Capsule U Surveillance Charpy Impact Data for Weld Metal (Wire Heat 25531 / Flux Lot 1211) - Refitted Using Hyperbolic Tangent Curve-Fitting Method -





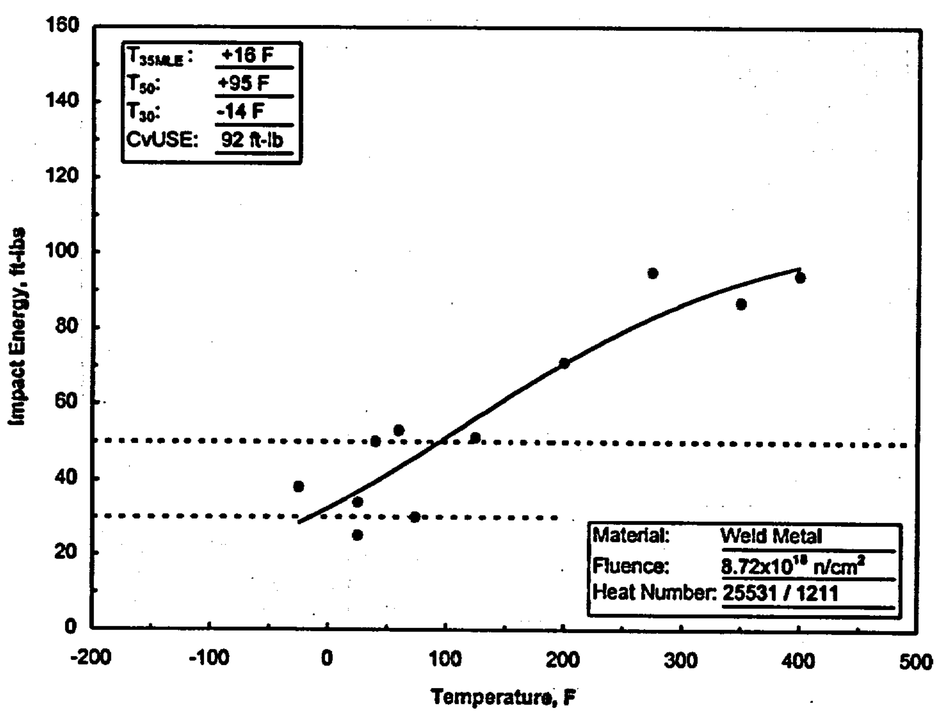


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Figure D-10. Unirradiated Surveillance Charpy V-Notch Impact Data for North Anna Unit No. 1, Heat-Affected-Zone Material - Refitted Using Hyperbolic Tangent Curve-Fitting Method -

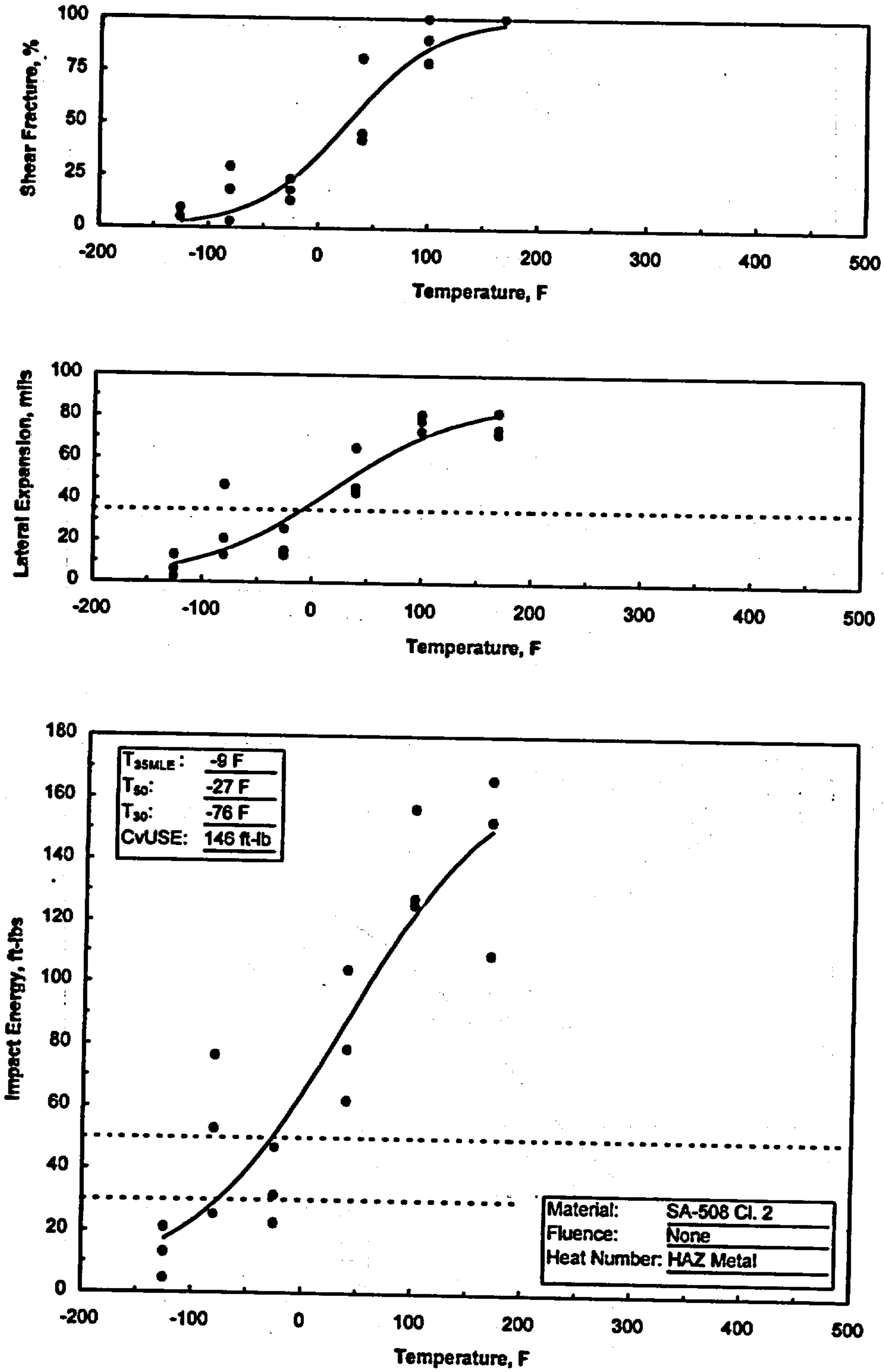

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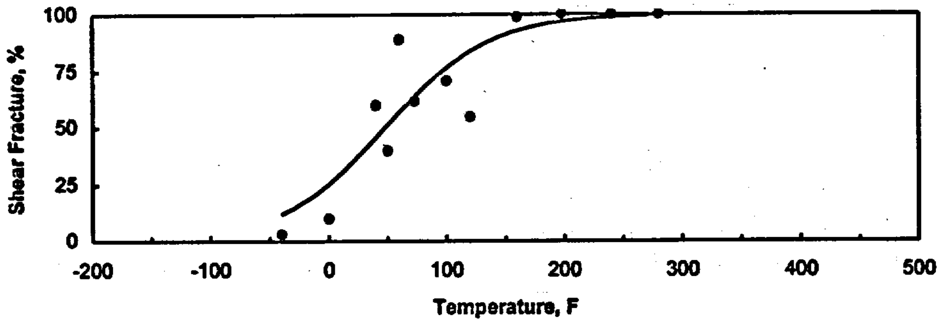
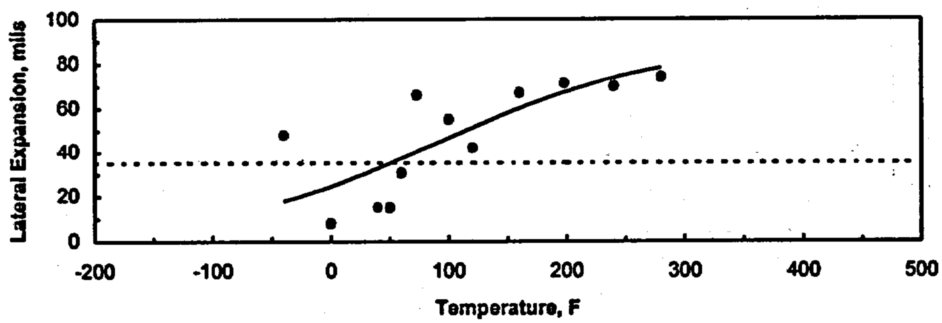
Figure D-11. North Anna Unit No. 1 Capsule V Surveillance Charpy Impact Data for Heat-Affected-Zone Material
- Refitted Using Hyperbolic Tangent Curve-Fitting Method -





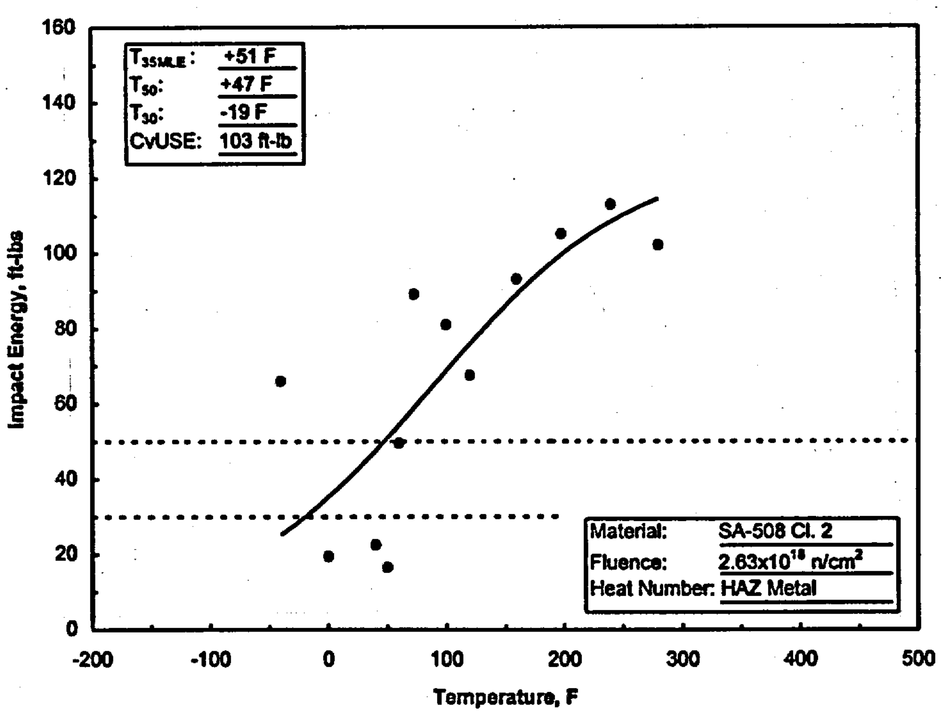
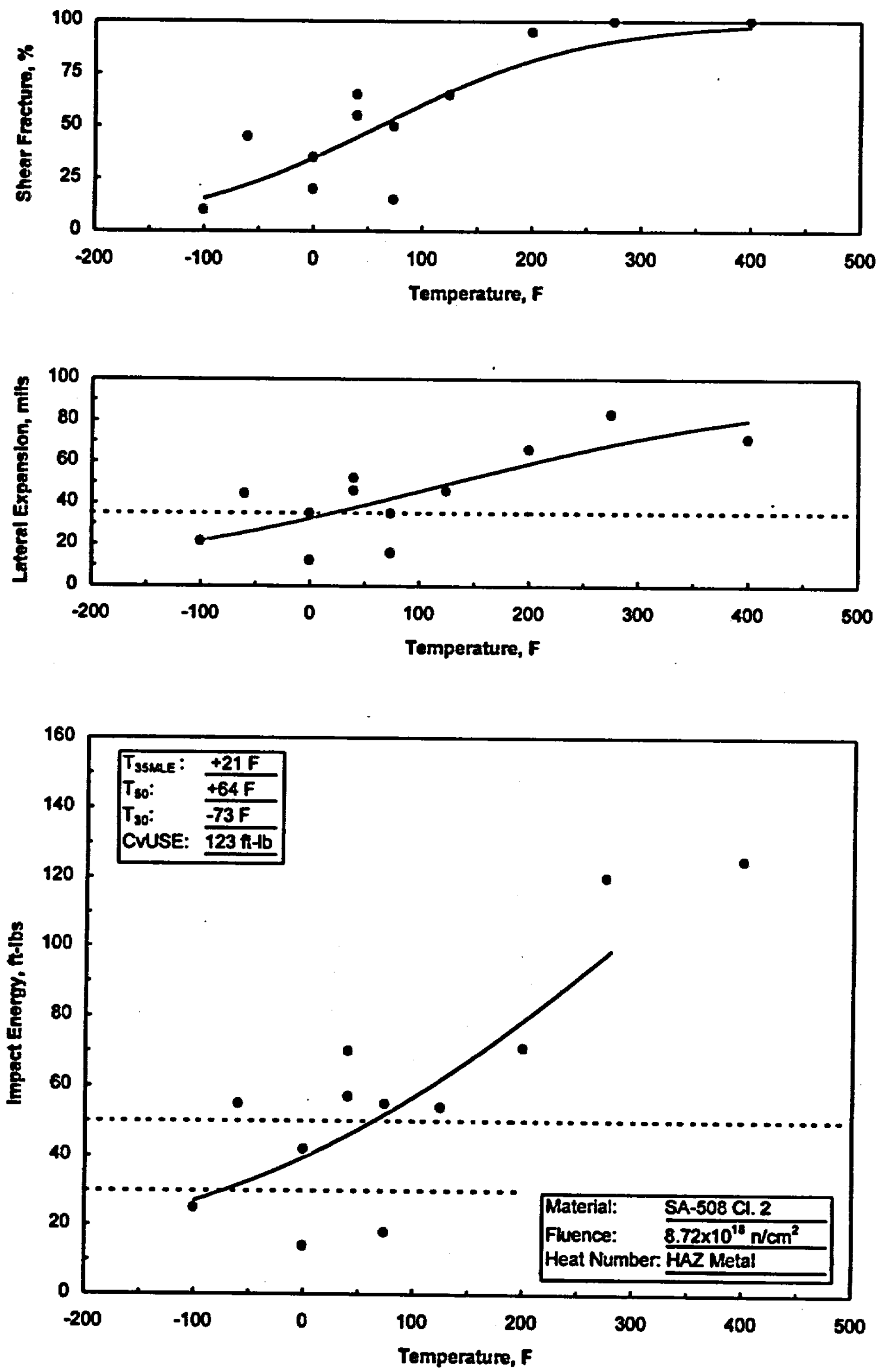


Figure D-12. North Anna Unit No. 1 Capsule U Surveillance Charpy Impact Data for Heat-Affected Zone Material
- Refitted Using Hyperbolic Tangent Curve-Fitting Method -



FRAMATOME TECHNOLOGIES

APPENDIX E

Charpy V-Notch Shift Comparison: Hand-Drawn Curve Fitting vs. Hyperbolic Tangent Curve Fitting

Table E-1. Comparison of Curve Fit Transition Temperature Shifts for North Anna Unit No. 1 Surveillance Material, Base Metal Forging 03, Heat No. 990400/292332, Tangential Orientation

Capsule	Fluence ($x10^{19}$ n/cm^2) (E>1.0 MeV)	30 ft-lb Transition Temperature			
		Hand-Drawn Curve Fit		Hyperbolic Tangent Curve Fit	
		Avg., °F	Shift, °F	Avg., °F	Shift, °F
Unirradiated	---	-6	---	-5	---
V	0.263	33	39	46	51
U	0.872	89	95	111	116

Capsule	Fluence ($x10^{19}$ n/cm^2) (E>1.0 MeV)	50 ft-lb Transition Temperature			
		Hand-Drawn Curve Fit		Hyperbolic Tangent Curve Fit	
		Avg., °F	Shift, °F	Avg., °F	Shift, °F
Unirradiated	---	12	---	23	---
V	0.263	80	68	84	61
U	0.872	112	100	145	122

Capsule	Fluence ($x10^{19}$ n/cm^2) (E>1.0 MeV)	35 MLE Transition Temperature			
		Hand-Drawn Curve Fit		Hyperbolic Tangent Curve Fit	
		Avg., °F	Shift, °F	Avg., °F	Shift, °F
Unirradiated	---	7	---	17	---
V	0.263	80	73	78	61
U	0.872	112	105	122	105

Table E-2. Comparison of Curve Fit Transition Temperature Shifts for North Anna Unit No. 1 Surveillance Material, Base Metal Forging 03, Heat No. 990400/292332, Axial Orientation

Capsule	Fluence ($x10^{19}$ n/cm^2) (E>1.0 MeV)	30 ft-lb Transition Temperature			
		Hand-Drawn Curve Fit		Hyperbolic Tangent Curve Fit	
		Avg., °F	Shift, °F	Avg., °F	Shift, °F
Unirradiated	---	46	---	40	---
V	0.263	65	19	69	29
U	0.872	111	65	112	72

Capsule	Fluence ($x10^{19}$ n/cm^2) (E>1.0 MeV)	50 ft-lb Transition Temperature			
		Hand-Drawn Curve Fit		Hyperbolic Tangent Curve Fit	
		Avg., °F	Shift, °F	Avg., °F	Shift, °F
Unirradiated	---	85	---	84	---
V	0.263	130	45	123	39
U	0.872	165	80	165	81

Capsule	Fluence ($x10^{19}$ n/cm^2) (E>1.0 MeV)	35 MLE Transition Temperature			
		Hand-Drawn Curve Fit		Hyperbolic Tangent Curve Fit	
		Avg., °F	Shift, °F	Avg., °F	Shift, °F
Unirradiated	---	74	---	72	---
V	0.263	110	36	106	34
U	0.872	124	50	129	57

Table E-3. Comparison of Curve Fit Transition Temperature Shifts for North Anna Unit No. 1 Surveillance Material, Weld Metal (Wire Heat 25531 / Flux Lot 1211)

Capsule	Fluence ($x10^{19}$ n/cm^2) (E>1.0 MeV)	30 ft-lb Transition Temperature			
		Hand-Drawn Curve Fit		Hyperbolic Tangent Curve Fit	
		Avg., °F	Shift, °F	Avg., °F	Shift, °F
Unirradiated	---	-26	---	-44	---
V	0.263	52	78	44	88
U	0.872	49	75	-14	30

Capsule	Fluence ($x10^{19}$ n/cm^2) (E>1.0 MeV)	50 ft-lb Transition Temperature			
		Hand-Drawn Curve Fit		Hyperbolic Tangent Curve Fit	
		Avg., °F	Shift, °F	Avg., °F	Shift, °F
Unirradiated	---	25	---	17	---
V	0.263	96	71	90	73
U	0.872	125	100	95	78

Capsule	Fluence ($x10^{19}$ n/cm^2) (E>1.0 MeV)	35 MLE Transition Temperature			
		Hand-Drawn Curve Fit		Hyperbolic Tangent Curve Fit	
		Avg., °F	Shift, °F	Avg., °F	Shift, °F
Unirradiated	---	-13	---	-26	---
V	0.263	67	80	60	86
U	0.872	52	65	16	42

Table E-4. Comparison of Curve Fit Transition Temperature Shifts for North Anna Unit No. 1 Surveillance Material, Heat-Affected-Zone Material

Capsule	Fluence ($x10^{19}$ n/cm^2) (E>1.0 MeV)	30 ft-lb Transition Temperature			
		Hand-Drawn Curve Fit		Hyperbolic Tangent Curve Fit	
		Avg., °F	Shift, °F	Avg., °F	Shift, °F
Unirradiated	---	-51	---	-76	---
V	0.263	4	55	-19	57
U	0.872	49	100	-73	3

Capsule	Fluence ($x10^{19}$ n/cm^2) (E>1.0 MeV)	50 ft-lb Transition Temperature			
		Hand-Drawn Curve Fit		Hyperbolic Tangent Curve Fit	
		Avg., °F	Shift, °F	Avg., °F	Shift, °F
Unirradiated	---	2	---	-27	---
V	0.263	60	58	47	74
U	0.872	112	110	64	91

Capsule	Fluence ($x10^{19}$ n/cm^2) (E>1.0 MeV)	35 MLE Transition Temperature			
		Hand-Drawn Curve Fit		Hyperbolic Tangent Curve Fit	
		Avg., °F	Shift, °F	Avg., °F	Shift, °F
Unirradiated	---	20	---	-9	---
V	0.263	65	45	51	60
U	0.872	90	70	21	30

APPENDIX F

Fluence Analysis Methodology

The primary tool used in the determination of the flux and fluence exposure to the Capsule W specimens is the two-dimensional discrete ordinates transport code DORT.[F-1]

The North Anna Unit No. 1 Capsule W was located at the 25° location for cycles 1 through 13. The power distributions in the cycle 1 - 13 irradiation were symmetric both in θ and Z. That is, the axial power shape is roughly the same for any angle and, conversely, that the azimuthal power shape is the same for any height. This means that the neutron flux at some point (R, θ, Z) can be considered to be a separable function of (R, θ) and (R, Z). Therefore, the cycle 1 - 13 irradiation can be modeled using the standard FTI synthesis procedures.[F-2]

Figure F-1 depicts the analytical procedure that is used to determine the fluence accumulated over cycles 1 - 13. As shown in the figure, the analysis is divided into seven tasks: (1) generation of the neutron source, (2) development of the DORT geometry models, (3) calculation of the macroscopic material cross sections, (4) synthesis of the results, and (5-7) estimation of the calculational bias, the calculational uncertainty, and the final fluence. Each of these tasks is discussed in greater detail in the following sections.

F.1. Generation of the Neutron Source

The time-average space and energy-dependent neutron source for cycles 1 - 13 was calculated using the SORREL[F-3] code. The effects of burnup on the spatial distribution of the neutron source were accounted for by calculating the cycle average fission spectrum for each fissile isotope on an assembly-by-assembly basis, and by determining the cycle-average specific neutron emission rate. This data was then used with the normalized time weighted average pin-by-pin relative power density (RPD) distribution to determine the space and energy-dependent neutron source. The azimuthal average, time average axial power shape in the peripheral assemblies was used with the fission spectrum of the peripheral assemblies to determine the neutron source for the axial DORT run. These two neutron source distributions were input to DORT as indicated in Figure F-1.

F.2. Development of the Geometrical Models

The system geometry models for the mid-plane (R, θ) DORT were developed using standard FTI interval size and configuration guidelines. The Rθ model for the cycle 1 -13 analysis extended radially from the center of the core to a point approximately 20 cm into the water of the shield tank, and azimuthally from the major axis to 45°. The surveillance capsule was modeled explicitly in the Rθ model and the axial (R, Z) DORT geometry model was developed using FTI procedures for axial modeling and the Virginia Power interval structure in the axial direction. The axial model extended from core plate to core plate. The geometrical models either met or exceeded all guidance criteria concerning interval size that are provided in U.S. NRC Draft Regulatory Guide DG-1053.[F-4] In all cases, cold dimensions were used. The geometry models were input to the DORT code as indicated in Figure F-1. These models will be used in all subsequent Code of Federal Regulation, Title 10, Part 50 (10 CFR 50), Appendix H[F-5] and pressure-temperature curve analyses that may be performed by FTI for North Anna Unit No. 1.

F.3. Calculation of Macroscopic Material Cross Sections

In accordance with DG-1053, the BUGLE-93[F-6] cross section library was used. The GIP[F-7] code was used to calculate the macroscopic energy-dependent cross sections for all materials used in the analysis – from the core out through the cavity and into the concrete and from core plate to core plate. The ENDF/B6 dosimeter reaction cross sections were used to generate the response functions that were used to calculate the DORT-calculated "saturated" specific activities.

F.4. DORT Analyses

The cross sections, geometry, and appropriate source were combined to create a set of DORT models (Rθ and RZ) for the cycle 1 - 13 analysis. Each DORT run utilized a cross section Legendre expansion of three (P_3), seventy directions (S_{10}) for the Rθ and forty-eight directions (S_8) for the RZ, and the appropriate boundary conditions. A theta-weighted flux extrapolation model was used, and all other requirements of DG-1053 that relate to the various DORT parameters were either met or exceeded for all DORT runs.

F.5. Synthesized Three Dimensional Results

The DORT analyses produce two sets of two-dimensional flux distributions, one for a vertical cylinder and one for the radial plane. The vertical cylinder, which will be referred to as the RZ plane, is defined as the plane bounded axially by the upper and lower grid plates and radially by the center of the core and a vertical line located 20 cm into the water biological shield. The horizontal plane, referred to as the Rθ plane, is defined as the plane bounded radially by the center of the core and a point located 20 cm into the water and azimuthally by the major axis and the adjacent 45° radius. The vessel flux, however, varies significantly in all three cylindrical-coordinate directions (R, θ, Z). This means that if a point of interest is outside the planes of both the R-Z DORT and the R-θ DORT, the true flux cannot be determined from either DORT run. Under the assumption that the three-dimensional flux is a separable function,[F-2] both two-dimensional data sets were mathematically combined to estimate the flux at all three-dimensional points (R, θ, Z) of interest. The synthesis procedure outlined in DG-1053 forms the basis for the FTI flux-synthesis process.

F.6. Calculated Activities and Measured Activities

The calculated activities for each dosimeter type "d" were determined using the following equation:

$$C_d = \sum_{g=1}^{G} \phi_g(\bar{r}_d) \times RF_g^d \times B_d \times NSF$$

where:

C_d	...	calculated specific activity for dosimeter "d" in μCi of product isotope per gram of target isotope
$\phi_g(\bar{r}_d)$	...	three dimensional flux for dosimeter "d" at position for energy group "g"
RF_g^d	...	dosimeter response function for dosimeter "d" and energy group "g"
B_d	...	bias correction factors for dosimeter "d"
NSF	...	non-saturation correction factor (NSF).

The bias correction factors (B_d) in the specific activity calculation above are listed in Table F-1.

The power history data in Table F-2 was used to determine the non-saturation factors for each of the dosimeter product isotopes for cycles 1 - 13.

A photofission factor was applied to correct for the fact that some of the ^{137}Cs atoms present in the dosimeter were produced by (γ, f) reactions and were not accounted for in DORT analysis. The short half life and impurity correction factors were insignificant and were not applied.

F.7. C/M Ratios

To start, the following explanation will define the meanings of the terms "measurements" (M) and "calculations" (C) as used in this analysis:[F-2]

- Measurements: The meaning of the term "measurements" as used by FTI is the measurement of the physical quantity of the dosimeter (specific activity) that responded to the neutron fluence, not to the "measured fluence." For the example of an iron dosimeter, a reference to the measurements means the specific activity of ^{54}Mn in μCi/g, which is the product isotope of the dosimeter reaction:

$$^{54}\mathrm{Fe} + {}^{1}_{0}\mathrm{n} \rightarrow {}^{54}\mathrm{Mn} + \mathrm{p}^{+}$$

- Calculations: The calculational methodology produces two primary results – the calculated dosimeter activities and the neutron flux at all points of interest. The meaning of the term "calculations" as used by FTI is the calculated dosimeter activity. The calculated activities are determined in such a way that they are directly comparable to the measurement values, but without recourse to the measurements. That is, the calculated values are determined by the DORT calculation and are directly comparable to the measurement values. ENDF/B6 based dosimeter reaction cross sections[F-8] and response functions were used in determining the calculated values for each individual dosimeter. In summary, it should be stressed that the calculation values in the FTI approach[1] are independent of the measurement values.

F.8. Uncertainty

The North Anna Unit No. 1 fluence predictions are based on the methodology described in the FTI "Fluence and Uncertainty Methodologies" topical report, BAW-2241P, Revision 1.[F-2] The time-averaged fluxes, and thereby the fluences throughout the reactor and vessel, are calculated with the DORT discrete ordinates computer code using three-dimensional synthesis methods. The basic theory for synthesis is described in Section 3.0 of BAW-2241P, Revision 1 and the DORT three-dimensional synthesis results are the bases for the fluence predictions using the FTI "Semi-Analytical" (calculational) methodology.

The uncertainties in the North Anna Unit No. 1 fluence values have been evaluated to ensure that the greater than 1.0 MeV calculated fluence values are accurate (with no discernible bias) and have a mean standard deviation that is consistent with the FTI benchmark database of uncertainties. Consistency between the fluence uncertainties in the updated calculations for North Anna Unit No. 1 cycles 1 through 13 and those in the FTI benchmark database ensures that the vessel fluence predictions are consistent with the 10 CFR 50.61,[F-9] Pressurized Thermal Shock (PTS) screening criteria and the Regulatory Guide 1.99, Revision 2[F-10] embrittlement evaluations.

The verification of the fluence uncertainty for the North Anna Unit 1 reactor includes:

- estimating the uncertainties in the cycles 1 - 13 dosimetry measurements,
- estimating the uncertainties in the cycles 1 - 13 benchmark comparison of calculations to measurements, and
- estimating the uncertainties in the cycles 1 - 13 pressure vessel fluence
- determining if the specific measurement and benchmark uncertainties for cycles 1 - 13 are consistent with the FTI database of generic uncertainties in the measurements and calculations.

The embrittlement evaluations in Regulatory Guide 1.99, Revision 2 and those in 10 CFR 50.61 for the PTS screening criteria apply a margin term to the reference temperatures. The margin term includes the product of a confidence factor of 2.0 and the mean embrittlement standard deviation. The factor of 2.0 implies a very high level of confidence in the fluence uncertainty as well as the uncertainty in the other variables contributing to the embrittlement.. The 12 dosimeter measurements from the North Anna Unit No. 1 Capsule W analysis would not directly support this high level of confidence. However, the North Anna Unit No. 1 Capsule W dosimeter measurement uncertainties are consistent with the FTI database. Therefore, the calculational uncertainties in the updated fluence predictions for North Anna Unit No. 1 are supported by 728 additional dosimeter measurements and thirty-nine benchmark comparisons of calculations to measurements as shown in Appendix A of BAW-2241P, Revision 1. The calculational uncertainties are also supported by the fluence sensitivity evaluation of the uncertainties in the physical and operational parameters, which are included in the vessel fluence uncertainty.[F-2] The dosimetry measurements and benchmarks, as well as the fluence sensitivity analyses in the topical are sufficient to support a 95 percent confidence level, with a confidence factor of ±2.0, in the fluence results from the "Semi-Analytical" methodology.

The FTI generic uncertainty in the capsule dosimetry measurements has been determined to be unbiased and has an estimated standard deviation of 7.0 percent for the qualified set of dosimeters. The North Anna Unit No. 1 cycle 1 - 13 dosimetry measurement uncertainties were evaluated to determine if any biases were evident and to estimate the standard deviation. The dosimetry measurements were found to be appropriately calibrated to standards traceable to the National Institute of Standards and Technology and are thereby unbiased by definition. The mean measurement uncertainties associated with cycle 1 - 13 are as follows:

Cycle	σ_M (%)
1 – 13	5.06

This value was determined from Equation 7.6 in BAW-2241P, Revision 1 and indicate that there is consistency with the FTI database. Consequently, when the FTI database is updated, the North Anna Unit No. 1 cycle 1 - 13 dosimetry measurement uncertainties may be combined with the other 728 dosimeters. Since the cycle 1 - 13 measurements are consistent with the FTI database, it is estimated that North Anna Unit No. 1 dosimeter measurement uncertainty may be represented by the FTI database standard deviation of 7.0 percent. Based on the FTI database, there appears to be a 95 percent level of confidence that 95 percent of the North Anna Unit No. 1 Capsule W dosimetry measurements, for fluence reactions above 1.0 MeV, are within ±14.2 percent of the true values.

The FTI generic uncertainty for benchmark comparisons of capsule dosimetry calculations relative to the measurements indicates that any benchmark bias in the greater than 1.0 MeV results is too small to be uniquely identified. The estimated standard deviation between the calculations and measurements is 9.9 percent. This implies that the root mean square deviation between the FTI calculations of the North Anna Unit No. 1 dosimetry and the measurements should be approximately 9.9 percent in general and bounded by ±20.04 percent for a 95 percent confidence interval with thirty-nine independent benchmarks.

The weighted mean values of the ratio of calculated dosimeter activities to measurements (C/M) for North Anna Unit No. 1 Capsule W have been statistically evaluated using Equation 7.15 from BAW-2241P, Revision 1. The standard deviations in the benchmark comparisons are as follows:

Cycle	$\sigma_{C/M}$ (%)
1 – 13	3.0

This standard deviation indicates that the benchmark comparisons are consistent with the FTI database. Consequently, when the FTI database is updated, the cycle 1 - 13 benchmark uncertainties may be included with the other thirty-nine benchmark uncertainties in BAW-2241P, Revision 1. The consistency between the cycle 1 - 13 benchmark uncertainties and those in the FTI database indicates that North Anna Unit No. 1 fluence calculations for cycles 1 - 13 have no discernible bias in the greater than 1.0 MeV fluence values. In addition, the consistency indicates that the fluence values can be represented by the FTI reference set which includes a standard deviation of 7.0 percent at dosimetry locations. That is

$$\sigma_{\text{capsule fluence}} \leq 7.00\%$$

$$\sigma_{\text{pressure vessel fluence}} \leq 10.00\% .$$

Table F-1. Bias Correction Factors

Dosimeter Type	Bias
Activation	Short Half Life
Fission	Photofission Impurities

Table F-2. North Anna Unit 1 Monthly Power History

Cycle 1				
Month	Year	Average Power (MW)	Maximum Power (MW)	Relative Power
April	1978	347	2775	0.12505
May	1978	1141	2775	0.41117
June	1978	2303	2775	0.82991
July	1978	2353	2775	0.84793
August	1978	2633	2775	0.94883
September	1978	1995	2775	0.71892
October	1978	2350	2775	0.84685
November	1978	2631	2775	0.94811
December	1978	2584	2775	0.93117
January	1979	2298	2775	0.82811
February	1979	2384	2775	0.85910
March	1979	2478	2775	0.89297
April	1979	0	2775	0.00000
May	1979	2439	2775	0.87892
June	1979	2703	2775	0.97405
July	1979	2672	2775	0.96288
August	1979	2747	2775	0.98991
September	1979	1948	2775	0.70198
October	1979	0	2775	0.00000
November	1979	0	2775	0.00000
December	1979	0	2775	0.00000

Table F-2. (Cont'd.) North Anna Unit 1 Monthly Power History

Cycle 2				
Month	Year	Average Power (MW)	Maximum Power (MW)	Relative Power
January	1980	561	2775	0.20216
February	1980	2131	2775	0.76793
March	1980	2711	2775	0.97694
April	1980	2370	2775	0.85405
May	1980	1956	2775	0.70486
June	1980	1804	2775	0.65009
July	1980	2631	2775	0.94811
August	1980	2739	2775	0.98703
September	1980	2622	2775	0.94486
October	1980	2475	2775	0.89189
November	1980	2492	2775	0.89802
December	1980	1851	2775	0.66703
January	1981	0	2775	0.00000
February	1981	0	2775	0.00000
March	1981	0	2775	0.00000

Table F-2. (Cont'd.) North Anna Unit 1 Monthly Power History

Cycle 3				
Month	Year	Average Power (MW)	Maximum Power (MW)	Relative Power
April	1981	1931	2775	0.69586
May	1981	2769	2775	0.99784
June	1981	2686	2775	0.96793
July	1981	2631	2775	0.94811
August	1981	2564	2775	0.92396
September	1981	2681	2775	0.96613
October	1981	533	2775	0.19207
November	1981	2506	2775	0.90306
December	1981	2753	2775	0.99207
January	1982	2603	2775	0.93802
February	1982	2614	2775	0.94198
March	1982	2772	2775	0.99892
April	1982	2395	2775	0.86306
May	1982	1190	2775	0.42883
June	1982	0	2775	0.00000
July	1982	0	2775	0.00000
August	1982	0	2775	0.00000
September	1982	0	2775	0.00000
October	1982	0	2775	0.00000
November	1982	0	2775	0.00000

Table F-2. (Cont'd.) North Anna Unit 1 Monthly Power History

Cycle 4				
Month	Year	Average Power (MW)	Maximum Power (MW)	Relative Power
December	1982	6	2775	0.00216
January	1983	0	2775	0.00000
February	1983	0	2775	0.00000
March	1983	1374	2775	0.49514
April	1983	2769	2775	0.99784
May	1983	2287	2775	0.82414
June	1983	2620	2775	0.94414
July	1983	2484	2775	0.89514
August	1983	2756	2775	0.99315
September	1983	2559	2775	0.92216
October	1983	616	2775	0.22198
November	1983	2595	2775	0.93514
December	1983	2769	2775	0.99784
January	1984	724	2775	0.26090
February	1984	1859	2775	0.66991
March	1984	2767	2775	0.99712
April	1984	2769	2775	0.99784
May	1984	2639	2775	0.95099
June	1984	0	2775	0.00000
July	1984	0	2775	0.00000
August	1984	0	2775	0.00000

Table F-2. (Cont'd.) North Anna Unit 1 Monthly Power History

Cycle 5				
Month	Year	Average Power (MW)	Maximum Power (MW)	Relative Power
September	1984	264	2775	0.09514
October	1984	2173	2775	0.78306
November	1984	2148	2775	0.77405
December	1984	2700	2775	0.97297
January	1985	2661	2775	0.95892
February	1985	2706	2775	0.97514
March	1985	2459	2775	0.88613
April	1985	2769	2775	0.99784
May	1985	2775	2775	1.00000
June	1985	2750	2775	0.99099
July	1985	2775	2775	1.00000
August	1985	1277	2775	0.46018
September	1985	1843	2775	0.66414
October	1985	2448	2775	0.88216
November	1985	2656	2775	0.95712

Table F-2. (Cont'd.) North Anna Unit 1 Monthly Power History

Cycle 6				
Month	Year	Average Power (MW)	Maximum Power (MW)	Relative Power
December	1985	422	2775	0.15207
January	1986	1091	2775	0.39315
February	1986	2534	2775	0.91315
March	1986	2581	2775	0.93009
April	1986	2769	2775	0.99784
May	1986	2589	2775	0.93297
June	1986	2567	2775	0.92505
July	1986	2772	2775	0.99892
August	1986	1657	2794	0.59306
September	1986	281	2893	0.09713
October	1986	2757	2893	0.95299
November	1986	2780	2893	0.96094
December	1986	2766	2893	0.95610
January	1987	2890	2893	0.99896
February	1987	2890	2893	0.99896
March	1987	2881	2893	0.99585
April	1987	2595	2893	0.89699
May	1987	0	2893	0.00000

Table F-2. (Cont'd.) North Anna Unit 1 Monthly Power History

Cycle 7				
Month	Year	Average Power (MW)	Maximum Power (MW)	Relative Power
June	1987	278	2893	0.09609
July	1987	518	2893	0.17905
August	1987	0	2893	0.00000
September	1987	0	2893	0.00000
October	1987	772	2893	0.26685
November	1987	2132	2893	0.73695
December	1987	1987	2893	0.68683
January	1988	668	2893	0.23090
February	1988	1915	2893	0.66194
March	1988	2245	2893	0.77601
April	1988	2890	2893	0.99896
May	1988	2890	2893	0.99896
June	1988	2873	2893	0.99309
July	1988	2861	2893	0.98894
August	1988	2335	2893	0.80712
September	1988	2890	2893	0.99896
October	1988	2838	2893	0.98099
November	1988	2893	2893	1.00000
December	1988	2745	2893	0.94884
January	1989	2821	2893	0.97511
February	1989	2297	2893	0.79399
March	1989	0	2893	0.00000
April	1989	0	2893	0.00000
May	1989	0	2893	0.00000
June	1989	0	2893	0.00000

Table F-2. (Cont'd.) North Anna Unit 1 Monthly Power History

Cycle 8				
Month	Year	Average Power (MW)	Maximum Power (MW)	Relative Power
July	1989	2074	2893	0.71690
August	1989	2893	2893	1.00000
September	1989	2890	2893	0.99896
October	1989	2890	2893	0.99896
November	1989	2867	2893	0.99101
December	1989	1357	2893	0.46906
January	1990	2717	2893	0.93916
February	1990	2890	2893	0.99896
March	1990	2893	2893	1.00000
April	1990	2893	2893	1.00000
May	1990	2870	2893	0.99205
June	1990	2887	2893	0.99793
July	1990	2890	2893	0.99896
August	1990	2887	2893	0.99793
September	1990	2867	2893	0.99101
October	1990	2430	2893	0.83996
November	1990	1941	2893	0.67093
December	1990	1501	2893	0.51884
January	1991	1244	2893	0.43000
February	1991	0	2893	0.00000

Table F-2. (Cont'd.) North Anna Unit 1 Monthly Power History

Cycle 9				
Month	Year	Average Power (MW)	Maximum Power (MW)	Relative Power
March	1991	2419	2893	0.83616
April	1991	2881	2893	0.99585
May	1991	1374	2893	0.47494
June	1991	2890	2893	0.99896
July	1991	1533	2893	0.52990
August	1991	2725	2893	0.94193
September	1991	2887	2893	0.99793
October	1991	2890	2893	0.99896
November	1991	2893	2893	1.00000
December	1991	2123	2893	0.73384
January	1992	0	2893	0.00000
February	1992	0	2893	0.00000
March	1992	2109	2893	0.72900
April	1992	2728	2893	0.94297
May	1992	2745	2893	0.94884
June	1992	2745	2893	0.94884
July	1992	2725	2893	0.94193
August	1992	2728	2893	0.94297
September	1992	2583	2893	0.89284
October	1992	2106	2893	0.72796
November	1992	1678	2893	0.58002
December	1992	1319	2893	0.45593
January	1993	1131	2893	0.39094
February	1993	0	2893	0.00000
March	1993	0	2893	0.00000

Table F-2. (Cont'd.) North Anna Unit 1 Monthly Power History

Cycle 10				
Month	Year	Average Power (MW)	Maximum Power (MW)	Relative Power
April	1993	2309	2893	0.79813
May	1993	2893	2893	1.00000
June	1993	2890	2893	0.99896
July	1993	2890	2893	0.99896
August	1993	2893	2893	1.00000
September	1993	2893	2893	1.00000
October	1993	2890	2893	0.99896
November	1993	2893	2893	1.00000
December	1993	2893	2893	1.00000
January	1994	2893	2893	1.00000
February	1994	2893	2893	1.00000
March	1994	2893	2893	1.00000
April	1994	2890	2893	0.99896
May	1994	2893	2893	1.00000
June	1994	2867	2893	0.99101
July	1994	2447	2893	0.84583
August	1994	1950	2893	0.67404
September	1994	1661	2893	0.57414

Table F-2. (Cont'd.) North Anna Unit 1 Monthly Power History

Cycle 11				
Month	Year	Average Power (MW)	Maximum Power (MW)	Relative Power
October	1994	2471	2893	0.85413
November	1994	2890	2893	0.99896
December	1994	2893	2893	1.00000
January	1995	2722	2893	0.94089
February	1995	2893	2893	1.00000
March	1995	2873	2893	0.99309
April	1995	2893	2893	1.00000
May	1995	2893	2893	1.00000
June	1995	2893	2893	1.00000
July	1995	2890	2893	0.99896
August	1995	2893	2893	1.00000
September	1995	2893	2893	1.00000
October	1995	2893	2893	1.00000
November	1995	2893	2893	1.00000
December	1995	2803	2893	0.96889
January	1996	2317	2893	0.80090
February	1996	1987	2893	0.68683

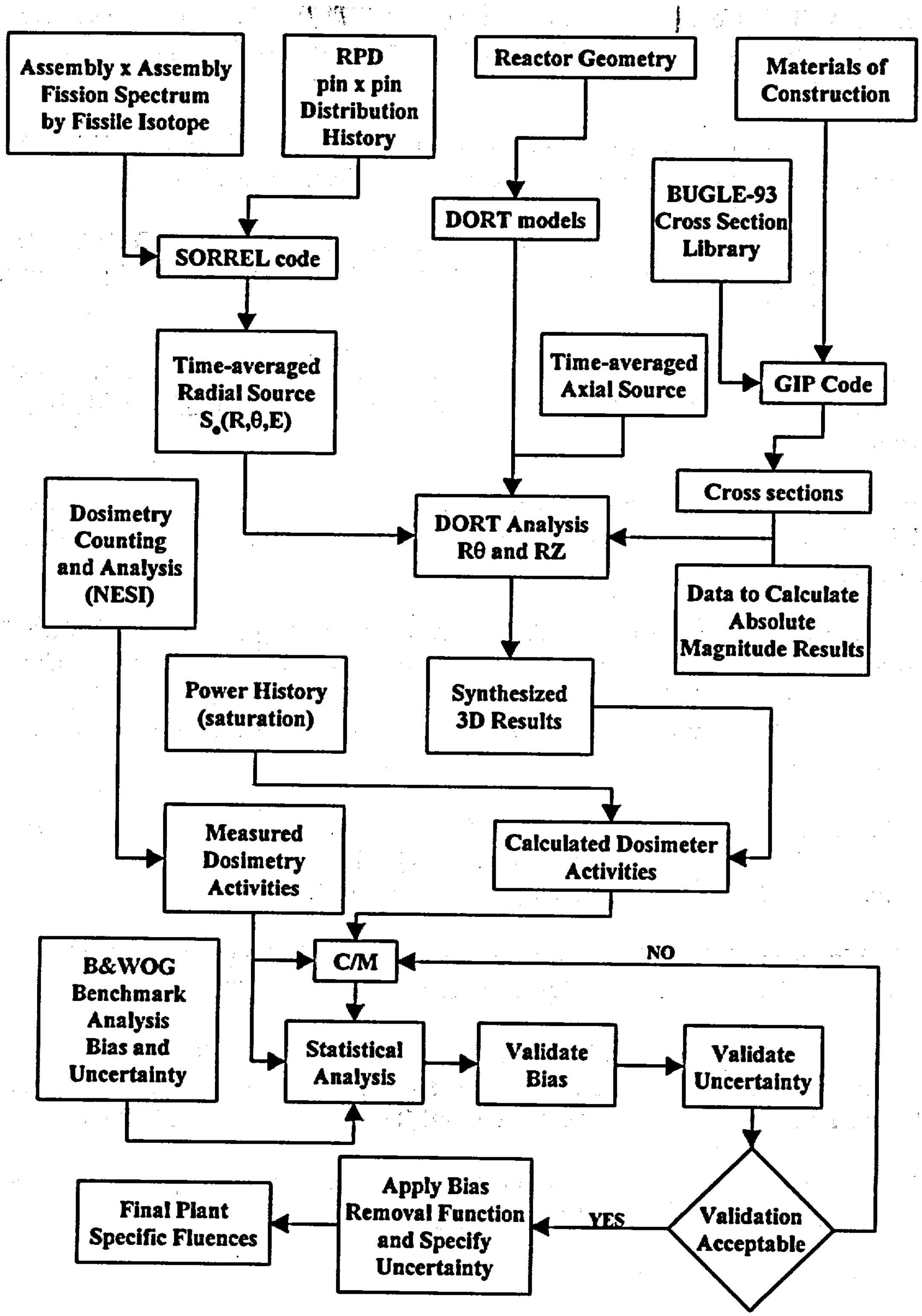
Table F-2. (Cont'd.) North Anna Unit 1 Monthly Power History

Cycle 12				
Month	Year	Average Power (MW)	Maximum Power (MW)	Relative Power
March	1996	2572	2893	0.88904
April	1996	2893	2893	1.00000
May	1996	2893	2893	1.00000
June	1996	2893	2893	1.00000
July	1996	2893	2893	1.00000
August	1996	2685	2893	0.92810
September	1996	2893	2893	1.00000
October	1996	2745	2893	0.94884
November	1996	2893	2893	1.00000
December	1996	2893	2893	1.00000
January	1997	2893	2893	1.00000
February	1997	2893	2893	1.00000
March	1997	2893	2893	1.00000
April	1997	2893	2893	1.00000
May	1997	2855	2893	0.98686

Table F-2. (Cont'd.) North Anna Unit 1 Monthly Power History

Cycle 13				
Month	Year	Average Power (MW)	Maximum Power (MW)	Relative Power
June	1997	2430	2893	0.83996
July	1997	2829	2893	0.97788
August	1997	2890	2893	0.99896
September	1997	2893	2893	1.00000
October	1997	2893	2893	1.00000
November	1997	2887	2893	0.99793
December	1997	2893	2893	1.00000
January	1998	2893	2893	1.00000
February	1998	2803	2893	0.96889
March	1998	2890	2893	0.99896
April	1998	2893	2893	1.00000
May	1998	2893	2893	1.00000
June	1998	2879	2893	0.99516
July	1998	2612	2893	0.90287
August	1998	2893	2893	1.00000

Figure F-1. Fluence Analysis Methodology for North Anna Unit No. 1 Capsule W



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F.9. References

F-1. Ed. M. A. Rutherford, N. M. Hassan, et al., *"DORT, Two Dimensional Discrete Ordinates Transport Code,"* BWNT-TM-107, Framatome Technologies, Inc., Lynchburg, Virginia, May 1995.

F-2. J. R. Worsham III, *"Fluence and Uncertainity Methodologies,"* BAW-2241P, Revision 1, Framatome Technologies, Inc., Lynchburg, Virginia, April 1999.

F-3. L. A. Hassler and N. M. Hassan, *"SORREL, DOT Input Generation Code User's Manual,"* NPGD-TM-427, Revision 8, Framatome Technologies, Inc., Lynchburg, Virginia, July 1992.

F-4. U.S. Nuclear Regulatory Commission, *"Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence,"* Draft Regulatory Guide DB-1053, June 1996.

F-5. Code of Federal Regulation, Title 10, Part 50, *"Domestic Licensing of Production and Utilization Facilities,"* Appendix H, Reactor Vessel Material Surveillance Program Requirements.

F-6. D. T. Ingersoll, et al., *"BUGLE-93, Production and Testing of the VITAMIN-B6 Fine Group and the BUGLE-93 Broad Group Neutron/Photon Cross-Section Libraries Derived from ENDF/B-VI Nuclear Data,"* ORNL-DLC-175, Radiation Safety Information Computational Center, Oak Ridge National Laboratory, Oak Ridge, Tennessee, April 1994.

F-7. L. A. Hassler and N. M. Hassan, *"GIP Users Manual for B&W Version, Group Organized Cross Section Input Program,"* NPGD-TM-456, Revision 11, Framatome Technologies, Inc., Lynchburg, Virginia, August 1994.

F-8. J. R. Worsham, *"BUGLE-93 Response Functions,"* FTI Document 32-1232719-00, Framatome Technologies, Inc., Lynchburg, Virginia, June 1995.

F-9. Code of Federal Regulations, Title 10, *"Domestic Licensing of Production and Utilization Facilities,"* Part 50.61, *"Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock,"* Federal Register, December 19, 1995.

F-10. U.S. Nuclear Regulatory Commission, *"Radiation Embrittlement of Reactor Vessel Materials,"* Regulatory Guide 1.99, Revision 2, May 1998.