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Enclosure 4

NON-PROPRIETARY REFERENCE DOCUMENTS AND A REDACTED VERSION OF A PROPRIETARY REFERENCE DOCUMENT (PUBLIC VERSION)

Virgil C. Summer (VCSNS) Unit 1
Dominion Energy South Carolina, Inc. (DESC)

Enclosure 4
Attachment 1

PWROG-21037-NP, REVISION 2

Virgil C. Summer (VCSNS) Unit 1
Dominion Energy South Carolina, Inc. (DESC)

PRESSURIZED WATER REACTOR OWNERS GROUP

PWROG
PWR Owners Group

PWROG-21037-NP
Revision 2

WESTINGHOUSE NON-PROPRIETARY CLASS 3

Determination of Unirradiated RT_{NDT} and Upper-Shelf Energy Values of the V.C. Summer Unit 1 Reactor Vessel Materials

Materials Committee

PA-MSC-1367, Tasks 1-3

March 2023

Westinghouse

Westinghouse

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Date: March 3, 2023

Your ref: N/A
Our ref: PWROG-21037-NP, Revision 2

Subject: **Determination of Unirradiated RT_{NDT} and Upper-Shelf Energy Values of the V.C. Summer Unit 1 Reactor Vessel Materials**

Attachments:
A. Summary of Unirradiated RT_{NDT} and Upper-Shelf Energy Values of the V.C. Summer Unit 1 Reactor Vessel Materials
B. PA-MSC-1367, Tasks 1 – 3 Evaluations for V.C. Summer Unit 1 Plate and Forging Materials
C. PA-MSC-1367, Tasks 1 – 3 Evaluations for V.C. Summer Unit 1 Reactor Vessel Welds

References
1. PWR Owners Group Letter OG-21-94, Revision 0, "Electronic Endorsement for V.C. Summer Unit 1 to Participate in PA-MSC-1367R0 'Document/Reconcile/Define Basis for Reactor Vessel Material Initial RTNDT and USE Values' Cafeteria Tasks 1-3," dated August 29, 2021.

Attachment A contains a summary of the results and methodologies used in the determination of the unirradiated nil-ductility transition temperature (RT_{NDT}) and Upper-Shelf Energy (USE) values for the V.C. Summer Unit 1 reactor vessel materials included under PA-MSC-1367, Tasks 1 – 3. This attachment also compares previously documented unirradiated RT_{NDT} and USE values with those updated herein. In addition, summary tables containing the chemistry, initial RT_{NDT} and unirradiated USE values for all of the V.C. Summer Unit 1 reactor vessel materials are provided.

Attachment B and Attachment C document the data and calculations for the determination of the unirradiated RT_{NDT} and USE values for the V.C. Summer Unit 1 reactor vessel materials updated under PA-MSC-1367 Tasks 1 – 3. Data was obtained from Certified Material Test Reports (CMTRs), vessel fabrication files, and Reactor Vessel Surveillance Program baseline report. The evaluations use the methodologies of ASME Code Section III to determine unirradiated RT_{NDT} and USE values, as appropriate.

Revision 1 of this report was issued to correct an error in the reported Lower Shell (Heat # C9923-2) initial RT_{NDT} and USE values in Table A.4-2, "Summary of V.C. Summer Unit 1 Reactor Vessel Material Properties." The incorrect values are only in the summary table. Changes are marked with change bars.

Revision 2 of this report was issued to correct an error in the Table A.2-1 for the "Original RT_{NDT}" for the Nozzle Shell (Heat # C0123-2)", the identification of the vessel fabricator in Section C.2, and the SMAWs' maximum & average Initial RT_{NDT} values in Table C.2-2.

Please transmit this technical report to Beth Haluska at Dominion.

Do not hesitate to contact the undersigned if you have any questions regarding the contents of this technical report.

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Attachment A
Summary of Unirradiated RT_{NDT} and Upper-Shelf Energy Values of the V.C. Summer Unit 1 Reactor Vessel Materials

A.1 Introduction

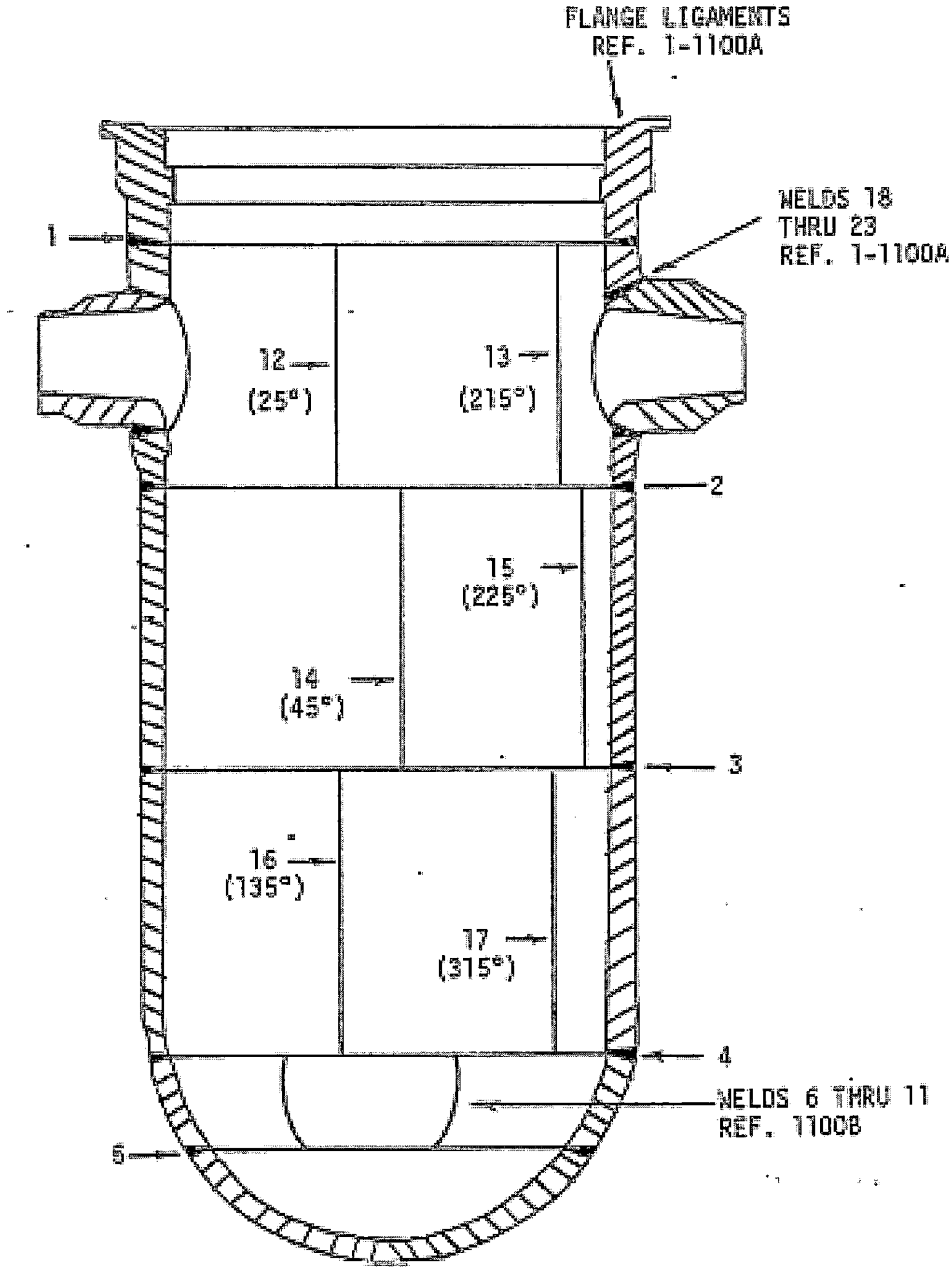
This attachment summarizes the results and methodologies used in the determination of the unirradiated nil-ductility transition temperature (RT_{NDT}) and Upper-Shelf Energy (USE) values for the V.C. Summer Unit 1 reactor vessel materials included under Dominion participation in Pressurized Water Reactor Owners Group (PWROG) Project Authorization (PA) PA-MSC-1367, Tasks 1 – 3 as documented in OG-21-94 [Ref. 1]. The material properties documented herein are based on all available data from the Westinghouse archives and the most up-to-date methodologies; therefore, these values may supersede any previously documented values. Based on the data available, the RT_{NDT} is determined using the methods in Sub-article NB-2331 of Section III of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (Code) [Ref. 2] or NUREG-0800, Branch Technical Position (BTP) 5-3 [Ref. 3]. The USE is determined using the methods American Society for Testing and Materials (ASTM) E185-82 [Ref. 4]. These methodologies used for determination of the initial material properties are detailed in this attachment.

Over time, NRC regulations and industry best practices for material property determination have changed. As a result of the age of V.C. Summer Unit 1, which began commercial operation in 1982, it is appropriate to re-evaluate the material properties for these plants. The currently licensed base metal material property values were determined many years ago without the use of modern analytical techniques and without consideration of all weld material. Typically, only the weld data for the core region welds were documented in the Certified Material Test Report (CMTR) [Ref. 5]; however, additional data may be available for select materials in a given plant's reactor vessel. The testing and reporting requirements were significantly less strict than core region materials. As a result, new estimation methods (such as those discussed herein) were developed, which allow for the determination of initial properties using limited data. This report utilizes the most up-to-date methodologies and all available data to determine the most appropriate initial property values for the V.C. Summer Unit 1 reactor vessel materials. The methodologies used for the evaluations contained herein are described in subsequent sections of this attachment.

Following is a list of the V.C. Summer Unit 1 materials included in PA-MSC-1367, Tasks 1 – 3, and a reactor vessel schematic diagram is contained in Figure A.1-1. The associated welds were also investigated for this report. For some welds in the presumed extended beltline, the heat number used in specific weld seams could not be identified. To address these situations, bounding or generic V.C. Summer weld properties were developed to be used anywhere the specific weld heat cannot be identified:

- Replacement Reactor Vessel Closure Head
- Reactor Vessel Flange Forging
- Inlet Nozzle Forgings
- Outlet Nozzle Forgings
- Upper Shell Plates
- Intermediate Shell Plates
- Lower Shell Plates
- Bottom Head Ring
- Bottom Head Dome

Figure A.1-1 V.C. Summer Unit 1 Reactor Vessel Schematic



A.2 Initial RT_{NDT} Determination

Subarticle NB-2331 of Section III of the ASME Code [Ref. 2] requires both drop-weight test data as well as Charpy V-notch (CVN) test data from transverse specimens for determination of unirradiated RT_{NDT} values. This methodology is copied below:

A.2.1 ASME Code Section III, NB-2331, "Material for Vessels"

Pressure-retaining materials for vessels, other than bolting, shall be tested as follows.

(a) Establish a reference temperature RT_{NDT}; this shall be done as follows:

(1) Determine a temperature T_{NDT} that is at or above the nil-ductility transition temperature by drop weight tests.

(2) At a temperature not greater than T_{NDT} + 60°F (T_{NDT} + 33°C), each specimen of the C_v test (NB-2321.2) shall exhibit at least 35 mils (0.89 mm) lateral expansion and not less than 50 ft-lb (68 J) absorbed energy. Retesting in accordance with NB-2350 is permitted. When these requirements are met, T_{NDT} is the reference temperature RT_{NDT}.

(3) In the event that the requirements of (2) above are not met, conduct additional C_v tests in groups of three specimens (NB-2321.2) to determine the temperature T_{Cv} at which they are met. In this case the reference temperature RT_{NDT} = T_{Cv} – 60°F (T_{Cv} – 33°C). Thus, the reference temperature RT_{NDT} is the higher of T_{NDT} and [T_{Cv} – 60°F (T_{Cv} – 33°C)].

(4) When a C_v test has not been performed at T_{NDT} + 60°F (T_{NDT} + 33°C), or when the C_v test at T_{NDT} + 60°F (T_{NDT} + 33°C) does not exhibit a minimum of 50 ft-lb (68 J) and 35 mils (0.89 mm) lateral expansion, a temperature representing a minimum of 50 ft-lb (68 J) and 35 mils (0.89 mm) lateral expansion may be obtained from a full C_v impact curves developed from the minimum data points of all the C_v tests performed.

(b) Apply the procedures of NB-2331(a) to NB-2331(b)(1), (2), and (3):

(1) the base material;

(2) the base material, the heat affected zone, and weld metal from the weld procedure qualification tests in accordance with NB-4330;

(3) the weld metal of NB-2431.

The appropriate test data necessary to use these Code requirements for determination of unirradiated RT_{NDT} values is not available for all V.C. Summer Unit 1 reactor vessel materials in the CMTR [Ref. 5]. As a result, the use of Branch Technical Position (BTP) 5-3 [Ref 3], formerly known as MTEB 5-2, was used in conjunction with Subarticle NB-2331 of Section III of the ASME Code for determination of initial RT_{NDT} for some reactor vessel materials, as necessary. Figure A.2-1 shows the orientation of the "strong" direction (longitudinal or tangential) test specimens compared to the "weak" direction (transverse or axial) test specimens. The BTP 5-3 methodology is copied below.

A.2.2 BTP 5-3 Methodology for Initial RT_{NDT} Determination as Documented in NUREG-0800

1.1 Determination of RT_{NDT} for Vessel Materials

Temperature limitations are determined in relation to a characteristic temperature of the material, RT_{NDT}, that is established from the results of fracture toughness tests. Both dropweight nil-ductility transition temperature (NDTT) [T_{NDT}] *tests and Charpy V-Notch tests should be run to determine the RT_{NDT}. The NDTT temperature, as determined by drop weight tests (ASTM E-208-1969) is the RT_{NDT} if, at 33°C (60°F) above the T_{NDT}, at least 68 J (50 ft-lbs) of energy and 0.89 mm (35 mils) lateral expansion (LE) are obtained in Charpy V-Notch tests on specimens oriented in the weak direction (transverse to the direction of maximum working).*

In most cases, the fracture toughness testing performed on vessel material for older plants did not include all tests necessary to determine the RT_{NDT} in this manner. Acceptable estimation methods for the most common cases, based on correlations of data from a large number of heats of vessel material, are provided below for guidance in determining RT_{NDT} when measured values are not available.

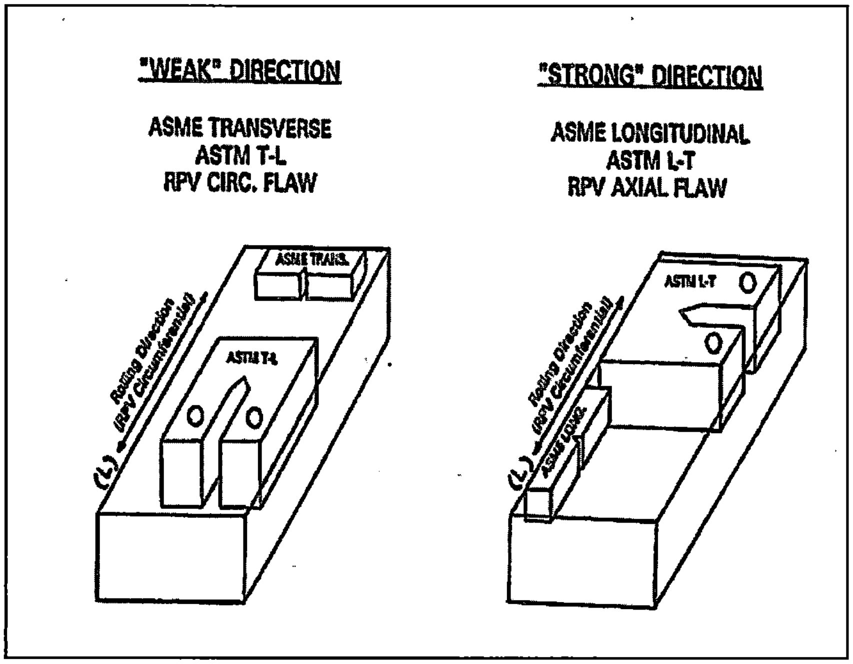
(1) If dropweight tests were not performed, but full Charpy V-notch curves were obtained, the NDTT for SA-533 Grade B, Class 1 plate and weld material may be assumed to be the temperature at which 41 J (30 ft-lbs) was obtained in Charpy V-notch tests, or -18°C (0°F), whichever was higher.

(2) If dropweight tests were not performed on SA-508, Class II forgings, the NDTT may be estimated as the lowest of the following temperatures:
   (a) 33°C (60°F).
   (b) The temperatures of the Charpy V-notch upper shelf
   (c) The temperature at which 136 J (100 ft-lbs) was obtained on Charpy V-notch tests if the upper shelf energy values were above 136 J (100 ft-lbs).

(3) If transversely-oriented Charpy V-notch specimens were not tested, the temperature at which 68 J (50 ft-lbs) and 0.89 mm (35 mils) lateral expansion would have been obtained on transverse specimens may be estimated by one of the following criteria:
   (a) Test results from longitudinally-oriented specimens reduced to 65% of their value to provide conservative estimates of values expected from transversely oriented specimens.
   (b) Temperatures at which 68 J (50 ft-lbs) and 0.89 mm (35 mils) LE were obtained on longitudinally-oriented specimens increased 11°C (20°F) to provide a conservative estimate of the temperature that would have been necessary to obtain the same values on transversely-oriented specimens.

(4) If limited Charpy V-notch tests were performed at a single temperature to confirm that at least 41 J (30 ft-lbs) was obtained, that temperature may be used as an estimate of the RT_{NDT} provided that at least 61 J (45 ft-lbs) was obtained if the specimens were longitudinally oriented. If the minimum value obtained was less than 61 J (45 ft-lbs), the RT_{NDT} may be estimated as 11°C (20°F) above the test temperature.

Figure A.2-1 Comparison of "Weak" Direction and "Strong" Direction Test Specimens



A.2.3 Methodology Implementation

Attachment B details the available drop-weight test data and Charpy V-notch test data for the V.C. Summer Unit 1 reactor vessel base metal materials included under PA-MSC-1367, Tasks 1 – 3. The data is used to determine the material properties, i.e., Cu, Ni, unirradiated RT_{NDT}, & unirradiated USE values. The unirradiated CVN test data were input into CVGRAPH, Version 6.02 for hyperbolic tangent curve fitting of the Charpy V-notch data sets, where appropriate. This software requires certain values to be fixed for the purposes of curve-fitting. Typically, this includes fixing the lower-shelf energy to 2.2 ft-lb for curve-fitting the CVN energy data. Similarly, the upper-shelf energy (USE) must also be fixed for curve-fitting the CVN energy data. USE calculation methodology is described in further detail in Section A.3. When graphing the minimum data points in accordance with ASME Code III Subarticle NB-2331 criteria, the upper-shelf value is fixed to the minimum absorbed energy or lateral expansion value with a shear ≥ 95%, regardless of the temperature at which it was achieved.

Attachment C details the available drop-weight test data and Charpy V-notch test data for the V.C. Summer Unit 1 reactor vessel weld materials included under PA-MSC-1367, Tasks 1 – 3. The data is used to evaluate material properties, i.e., Cu, Ni, unirradiated RT_{NDT}, & unirradiated USE, for all welds used in the fabrication of the V.C. Summer Unit 1 reactor vessel. These welds were identified from a review of CMTR [Ref. 5], Reactor Vessel Manual [Ref. 6], surveillance capsule program [Ref. 7], and Chicago Bridge & Iron (CB&I) fabrication records [Ref. 8] available to Westinghouse. Information from sister-plant Shearon Harris CMTR [Ref. 9] and surveillance capsule program [Ref. 10] were also used. The weld heats identified in the fabrication records were cross-referenced against an investigation performed on CB&I fabricated vessels to address NRC IE Bulletin 78-12 regarding atypical weld metal in reactor pressure vessel welds [Ref. 11]. In some situations, in the presumed extended beltline, the heat number used in specific weld seams could not be identified. To address these situations, a bounding RT_{NDT} value was identified based on the highest initial RT_{NDT} for all V.C. Summer weld heats that can be used anywhere the specific weld heat cannot be identified.

Table A.2-1 compares the initial RT_{NDT} values documented in the V.C. Summer FSAR with those updated herein. All material RT_{NDT} (both original and updated) were determined consistent with the NB-2300 section of the ASME Code Section III. The differences between the unirradiated RT_{NDT} values summarized in the V.C. Summer FSAR and those determined herein are a result of a change in curve-fitting method (hand-drawn versus hyperbolic tangent) used to fit the Charpy V-notch test data. Although some initial RT_{NDT} values increased as a result of the PA-MSC-1367 Tasks 1 – 3 evaluations, these updates do not require changes to the current V.C. Summer Pressure-Temperature (P-T) limit curves, as the updated materials remain non-limiting with respect to these curves. See Table A.2-2 for a more detailed disposition of the impact to the P-T limit curves.

Table A.2-1 V.C. Summer Unit 1 Initial RT_{NDT} Comparison for Materials Included Under PA-MSC-1367, Tasks 1 – 3

Material Identification	Original Initial RT_{NDT}[(a)] (°F)	Updated Initial RT_{NDT}[(b)] (°F)	Limiting Parameter[(c)] (°F)
Base Metals			
Replacement Reactor Vessel Closure Head (Heat # 2B145585 & 2B145586)	-34	-34	T_{NDT}
Vessel Flange (Heat # 5P5343, 4P4845, & 3P4570)	0	0	T_{NDT}
Inlet Nozzle 436B-1 (Heat # Q2Q41W)	-20	-20	T_{NDT}
Inlet Nozzle 436B-2 (Heat # Q2Q39W)	0	0	T_{NDT}
Inlet Nozzle 436B-3 (Heat # Q2Q39W)	-20	-20	T_{NDT}
Outlet Nozzle 437B-1 (Heat # Q2Q40)	-10	-10	T_{NDT}
Outlet Nozzle 437B-2 (Heat # Q2Q40W)	-10	-10	T_{NDT}
Outlet Nozzle 437B-3 (Heat # Q2Q44W)	0	0	T_{NDT}
Nozzle Shell (Heat # C9955-2)	18	9	$T_{50/35}$
Nozzle Shell (Heat # C0123-2)	26	15	$T_{50/35}$
Intermediate Shell (Heat # A9154-1)	30	21	$T_{50/35}$
Intermediate Shell (Heat # A9153-2)	-20	-20	T_{NDT}
Lower Shell (Heat # C9923-1)	10	5	$T_{50/35}$
Lower Shell (Heat # C9923-2)	10	4	$T_{50/35}$
Transition Ring (Heat # A9249-1)	-37	-40	T_{NDT}
Bottom Head (Heat # A9231-2)	-10	0[(d)]	$T_{50/35}$
Welds			
Nozzle to Intermediate Shell Circ. Weld (Heat # 4P4784, Flux Type Linde 124, Lot # 3930)	-44	-49	$T_{50/35}$
Intermediate Shell Long. Weld (Heat # 4P4784, Flux Type Linde 124, Lot # 3930)			
Intermediate to Lower Shell Circ. Weld (Heat # 4P4784, Flux Type Linde 124, Lot # 3930)			
Lower Shell Long. Weld (Heat # 4P4784, Flux Type Linde 124, Lot # 3930)			
Lower Shell to Transition Ring Circ. Weld (Heat # 3P4966, Flux Type Linde 124, Lot # 1214)	N/A	-20	T_{NDT}
Bounding Weld Value	N/A	10[(e)]	BTP 5-3

Notes for Table A.2-1:

(a) The original initial RT_{NDT} values were taken from the V.C. Summer FSAR, Table 5.2-11 [Ref. 12], as available. These values are consistent with those documented in WCAP-16305-NP [Ref. 13] and WCAP-16306-NP [Ref. 14].

(b) The evaluations of the updated initial RT_{NDT} values are detailed in Attachment B for base metal and Attachment C for welds.

(c) As documented in Attachments B and C, the limiting parameter is defined as the testing parameter that governs the initial RT_{NDT} determination. It can be one of the following values:

- Nil-Ductility Transition Temperature (T_{NDT}) determined per drop-weight testing.
- 50 ft-lb transition temperature ($T_{50\ ft\text{-}lb}$) based on measured transversely oriented specimen data sufficient to meet the ASME Code III Subarticle NB-2331 criteria.
- BTP 5-3 based on measured specimen data per BTP 5-3 Position 1.1(4).

(d) Based on strong-direction data, which is limiting.

(e) The bounding value based on the highest initial RT_{NDT} for all V.C. Summer weld heats.

Only one of the V.C. Summer Unit 1 materials had initial RT_{NDT} values that increased as a result of the technical evaluations performed under PA-MSC-1367 Tasks 1 – 3. However, this increase does not result in changes to the V.C. Summer P-T limit curves or in violations of the 10 CFR 50.61 [Ref. 15] Pressurized Thermal Shock (PTS) limits. Table A.2-2 summarizes the limiting one-quarter thickness (1/4T) and limiting three-quarter thickness (3/4T) adjusted reference temperature (ART) values applicable to the end of license renewal P-T limit curves taken from WCAP-16305-NP [Ref. 13]. The RT_{PTS} screening criteria values of 10 CFR 50.61 are 270°F for plates, forgings and axial weld materials, and 300°F for circumferential weld materials. The material with an initial RT_{NDT} value that increased are in regions of the reactor vessel not considered during the previous evaluations of the P-T limits and PTS; therefore, these increases have no adverse effect on the P-T limit curves or PTS analyses of record (AOR). The extended beltline consists of those materials which previously did not need to consider irradiation embrittlement but are projected to have fluence values greater than 1 x 10^{17} n/cm^2 at 72 EFPY, the criterion of RIS 2014-11 [Ref. 16] above which irradiation embrittlement needs to be considered. Table A.2-2 shows the effect on margin resulting from the updated initial RT_{NDT} value of the limiting material against the ART values documented in the P-T limits AOR and the 10 CFR 50.61 criteria. In future analyses, all materials with a projected fluence greater than 1 x 10^{17} n/cm^2 (E > 1.0 MeV) at the end of the licensed operating period should be considered in the P-T limit curves basis.

Table A.2-2 Available Margin Resulting from Initial RT_{NDT} Value Changes

Material / Reactor Vessel Integrity Limiting Value[a]	Intermediate Shell Plate (Heat # A9154-1)			
	AOR Initial RT_{NDT} Value[a] (°F)	**Updated Initial RT_{NDT} Value[b] (°F)**	**Added Margin[c] (°F)**	**Updated ART[d] (°F)**
1/4T ART – 153°F	30	21	9	144
3/4T ART – 138°F				129
RT_{PTS} – 159°F (Regulatory Limit 270°F)				150

Notes for Table A.2-2:

(a) AOR ART values are taken from WCAP-16305-NP [Ref. 13] and WCAP-16306-NP [Ref. 14].

(b) Taken from Table A.2-1.

(c) Added margin = AOR initial RT_{NDT} - the updated RT_{NDT} values.

(d) Updated ART = AOR limiting ART - added margin.

A.3 Upper-Shelf Energy (USE) Determination

The current 10 CFR 50, Appendix G [Ref. 17] requirements specify that upper-shelf energy (USE) be calculated based on ASTM E185-82. Herein, USE is calculated based on an interpretation of ASTM E185-82 that is best explained by the most recent version of the ASTM E185 manual (2016 version).

A.3.1 ASTM E185

ASTM E185-16, Section 3.1.5, defines the *Charpy upper-shelf energy level* as the following:

> *"the average energy value for all Charpy specimen tests (preferably three or more) whose test temperature is at or above the Charpy upper-shelf onset; specimens tested at temperatures greater than 83°C (150°F) above the Charpy upper-shelf onset shall not be included, unless no data are available between the onset temperature and onset +83°C (+150°F)."*

ASTM E185-16 [Ref. 18], Section 3.1.6, defines *Charpy upper-shelf onset* as the following:

> *"the temperature at which the fracture appearance of all Charpy specimens tested is at or above 95% shear."*

Using the guidelines in ASTM E185-82 and ASTM E185-16, the average of all Charpy data ≥ 95% shear is reported as the USE. In some instances, there may be data deemed 'out of family,' which are removed from the determination of the USE based on engineering judgment. However, the use of engineering judgment, per the standard Westinghouse methodology, to remove 'out of family' data was not necessary for V.C. Summer Unit 1.

A.3.2 Methodology Implementation

Details of the evaluation and re-evaluation performed to determine the USE values for the V.C. Summer Unit 1 materials included under PA-MSC-1367, Tasks 1 – 3 are contained in Attachment B and Attachment C for the base metals and welds, respectively. Comparisons of the updated unirradiated USE values from this evaluation to the USE values documented in the V.C. Summer FSAR are documented in Table A.3-1. In some situation, no data are available with shear ≥ 95%. In these cases, a minimum USE is provided based on the maximum shear experienced. To address the situations where specific weld wire heat numbers could not be identified for a specific weld seam, a generic USE value was defined as the mean minus two standard deviations of the USE values from all V.C. Summer weld heats with shear data ≥ 95%. The use of two standard deviations provides confidence that it will also bound those material with limited data, i.e., no shear data ≥ 95%.

Table A.3-1 V.C. Summer Unit 1 Initial USE Comparison for Materials Included Under PA-MSC-1367, Tasks 1 – 3

Material Identification	Original Unirradiated USE[(a)] (ft-lb)	Updated Unirradiated USE[(b)] (ft-lb)
Base Metals		
Replacement Reactor Vessel Closure Head (Heat # 2B145585 & 2B145586)	195.5	203
Vessel Flange (Heat # 5P5343, 4P4845, & 3P4570)	172	159
Inlet Nozzle 436B-1 (Heat # Q2Q41W)	130	152
Inlet Nozzle 436B-2 (Heat # Q2Q39W)	114.5[(c)]	115[(c)]
Inlet Nozzle 436B-3 (Heat # Q2Q39W)	135	138
Outlet Nozzle 437B-1 (Heat # Q2Q40)	146	159
Outlet Nozzle 437B-2 (Heat # Q2Q40W)	165	165
Outlet Nozzle 437B-3 (Heat # Q2Q44W)	150	155
Nozzle Shell (Heat # C9955-2)	100.5	101
Nozzle Shell (Heat # C0123-2)	91	91
Intermediate Shell (Heat # A9154-1)	80.5	76
Intermediate Shell (Heat # A9153-2)	106.5	107
Lower Shell (Heat # C9923-1)	106	106
Lower Shell (Heat # C9923-2)	91.5	92
Transition Ring (Heat # A9249-1)	107	107
Bottom Head (Heat # A9231-2)	134	125[(c)]
Welds		
Nozzle to Intermediate Shell Circ. Weld (Heat # 4P4784, Flux Type Linde 124, Lot # 3930)	84	86
Intermediate Shell Long. Weld (Heat # 4P4784, Flux Type Linde 124, Lot # 3930)		
Intermediate to Lower Shell Circ. Weld (Heat # 4P4784, Flux Type Linde 124, Lot # 3930)		
Lower Shell Long. Weld (Heat # 4P4784, Flux Type Linde 124, Lot # 3930)		
Lower Shell to Transition Ring Circ. Weld (Heat # 3P4966, Flux Type Linde 124, Lot # 1214)	-	> 88 ft-lb @ 85% Shear
Generic Weld Value	-	80[(d)]

Notes for Table A.3-1:

(a) The original initial RT_{NDT} values were taken from the V.C. Summer FSAR, Table 5.2-11 [Ref. 12], as available.

(b) The evaluations of the updated initial RT_{NDT} values are detailed in Attachment B for base metal and Attachment C for welds. USE values preceded by a greater than or equal to symbol, ">", identifies a material with no shear data ≥ 95%; thus, the initial USE values for these materials were set to greater than the impact energy with highest shear. The percent value identifies the shear value corresponding to the lower bound USE. These data points are excluded from the statistical analysis.

(c) Based on strong-direction data, which is limiting.

(d) The bounding USE value was defined as the mean minus two standard deviations of the USE values from all V.C. Summer weld heats with shear data ≥ 95%.

Some materials in V.C. Summer Unit 1, including the limiting Intermediate Shell (Heat # A9154-1), have initial USE values that decreased as a result of the technical evaluations performed under PA-MSC-1367 Tasks 1 – 3. However, these decreases do not result in violations of the USE screening criterion of 10 CFR 50, Appendix G [Ref. 17]. Table A.3-2 disposition the materials with decreased initial USE values evaluated in the AOR and shows that it remains above the 10 CFR 50, Appendix G criterion of 50 ft-lb for irradiated material.

It should be noted that the RAI responses [Ref. 19] and SER [Ref. 20] for V.C. Summer initial license renewal indicates a bounding USE for the Intermediate Shell of 51.75 ft-lb. However, this prediction is based solely on the initial USE of 75 ft-lb for the surveillance plate in WCAP-9234 [Ref. 7]. This is a more conservative number than the updated value in Table A.3-1 of 76 ft-lb. In addition, the projection in the AOR, WCAP-16306-NP, Appendix A, are based on the latest surveillance result from Capsule Z presented in WCAP-16298-NP [Ref. 21]. Regulatory Guide (RG) 1.99, Revision 2, [Ref. 22] states that the USE projections based on Position 2.2 should be used in preference to Position 1.2. RG 1.99, Position 2.2 is still used even though the surveillance data for Intermediate Shell (Heat # A9154-1) was non-credible. Credibility Criterion 3 of RG 1.99 indicates that even if the surveillance data are not considered credible for determination of ΔRT_{NDT}, "they may be credible for determining decrease in upper-shelf energy if the upper shelf can be clearly determined, following the definition given in ASTM E 185-82." Hence, the updated initial USE value for Intermediate Shell (Heat # A9154-1) will not cause it to drop below the 10 CFR 50, Appendix G limit.

Table A.3-2 10 CFR 50, Appendix G End of License Renewal USE Evaluation for Materials with Decreased Initial USE Values

Material Identification[(a)]	AOR Initial USE[(a)] (ft-lb)	Updated Initial USE[(b)] (ft-lb)	AOR USE Percent Decrease[(a)]	AOR Irradiated USE[(a)] (ft-lb)	Updated Irradiated USE[(c)] (ft-lb)	> 50 ft-lb? (Y/N)
Intermediate Shell Plate A9154-1	81	76	16%	68	64	Yes

Notes for Table A.3-2:

(a) AOR USE values are taken from WCAP-16306-NP, Appendix A [Ref. 14].

(b) Taken from Table A.3-1.

(c) Irradiated USE = Initial USE x (1 – Percent Decrease)

A.4 Material Properties Summary

As a part of the V.C. Summer Unit 1 participation in PA-MSC-1367, Tasks 1 – 3 [Ref. 1], the copper (Cu) and nickel (Ni) weight percent (wt. %) chemical compositions of the V.C. Summer Unit 1 reactor vessel materials not currently addressed by the 10 CFR 50, Appendix G, licensing basis were defined by a review of the available original test documentation. These reviews are contained in Attachment B and Attachment C for the base metals and welds, respectively. These materials include the V.C. Summer Unit 1 reactor vessel closure head, vessel flange, inlet and outlet nozzles, nozzle shell, bottom head ring and bottom head dome, and associated welds. When component specific chemistry data was not available, a generic value was defined as the mean plus one standard deviation of available data from similar materials. This method is consistent with RG 1.99, which allows the mean plus one standard deviation method to be used for conservative chemistry estimates based on generic data if component specific data is not available. The chemical compositions are summarized in Table A.4-2.

Some of data used to create the generic weld chemistry is included in plant surveillance programs. Specifically Heat # 4P4784, Flux Type Linde 124, Lot # 3930 is in the V.C. Summer surveillance program, documented in WCAP-9234 [Ref. 7], and Heat #5P6771, Flux Type Linde 124, Lot # 0342 is in the Shearon Harris surveillance program, documented in WCAP-10502 [Ref. 10]. The latest V.C. Summer surveillance data in WCAP-16298-NP [Ref. 21], Table 5-10, demonstrates that the measured ΔRT_{NDT} are less than RG 1.99, Position 1.1, predicted ΔRT_{NDT}. The latest Shearon Harris surveillance data in ANP-3798NP [Ref. 23], Table F-2, show measured ΔRT_{NDT} greater than RG 1.99, Position 1.1, predicted ΔRT_{NDT} at higher fluences. However, as shown in Table A.4-1, the RG 1.99, Position 1.1 chemistry factor (CF) based on the generic chemistry is greater than the RG 1.99, Position 2.1 (best fit), CF based on Shearon Harris surveillance data. Therefore, it is conservative to use the generic chemistry data to calculate RG 1.99, Position 1.1, CF.

Table A.4-1 Comparison of Chemistry Factors

	PWROG-21037-NP	Shearon Harris Surveillance Data	V.C. Summer Surveillance Data
CF	82[(a)]	72.6	42.2
Source	RG 1.99, Table 2	ANP-3798NP, Table F-4	WCAP-16305-NP, Table 3

Note for Table A.4-1:

(a) CF based on the generic weld chemistry data of Cu = 0.06% and Ni = 1.01% from Table A.4-2.

Table A.4-2 also summarizes the updated initial RT_{NDT} and unirradiated USE values for all of the V.C. Summer Unit 1 reactor vessel materials. Since Table A.4-2 is based on the most up-to-date analyses of the V.C. Summer Unit 1 reactor vessel materials, in future analyses, Table A.4-2 can be referenced for all V.C. Summer Unit 1 reactor vessel initial material properties.

Table A.4-2 Summary of V.C. Summer Unit 1 Reactor Vessel Material Properties

Material Identification	Wt. % Cu	Wt. % Ni	Initial RT_{NDT} (°F)	σ_I [(a)] (°F)	Unirradiated USE (ft-lb)
Base Metals					
Replacement Reactor Vessel Closure Head (Heat # 2B145585 & 2B145586)	0.03	0.73	-34	0	203
Vessel Flange (Heat # 5P5343, 4P4845, & 3P4570)	0.153[(c)]	0.70	0	0	159
Inlet Nozzle 436B-1 (Heat # Q2Q41W)	0.127[(b)]	0.76	-20	0	152
Inlet Nozzle 436B-2 (Heat # Q2Q39W)	0.127[(b)]	0.82	0	0	115
Inlet Nozzle 436B-3 (Heat # Q2Q39W)	0.127[(b)]	0.82	-20	0	138
Outlet Nozzle 437B-1 (Heat # Q2Q40)	0.127[(b)]	0.85	-10	0	159
Outlet Nozzle 437B-2 (Heat # Q2Q40W)	0.127[(b)]	0.80	-10	0	165
Outlet Nozzle 437B-3 (Heat # Q2Q44W)	0.127[(b)]	0.78	0	0	155
Nozzle Shell (Heat # C9955-2)	0.13	0.57	9	0	101
Nozzle Shell (Heat # C0123-2)	0.12	0.58	15	0	91
Intermediate Shell (Heat # A9154-1)	0.10	0.51	21	0	76
Intermediate Shell (Heat # A9153-2)	0.09	0.45	-20	0	107
Lower Shell (Heat # C9923-1)	0.08	0.41	5	0	106
Lower Shell (Heat # C9923-2)	0.08	0.41	4	0	92
Transition Ring (Heat # A9249-1)	0.172[(c)]	0.53	-40	0	107
Bottom Head (Heat # A9231-2)	0.172[(c)]	0.45	0	0	125
Welds					
Nozzle to Intermediate Shell Circ. Weld (Heat # 4P4784, Flux Type Linde 124, Lot # 3930)	0.05	0.91	-49	0	86
Intermediate Shell Long. Weld (Heat # 4P4784, Flux Type Linde 124, Lot # 3930)					
Intermediate to Lower Shell Circ. Weld (Heat # 4P4784, Flux Type Linde 124, Lot # 3930)					
Lower Shell Long. Weld (Heat # 4P4784, Flux Type 124, Lot # 3930)					
Lower Shell to Transition Ring Circ. Weld (Heat # 3P4966, Flux Type 124, Lot # 1214)	0.03	0.90	-20	0	> 88 ft-lb @ 85% Shear
Generic Weld Value	0.06[(d)]	1.01[(d)]	10	0	80

Notes for Table A.4-2:

(a) σ_I set equal to 0°F per WCAP-14040-A [Ref. 24] if the initial RT_{NDT} is based on ASME Code Section III / BTP 5-3 using measured data, unless otherwise noted.

(b) Generic value for SA-508 Class 2 nozzle forgings from PWROG-15109-NP-A [Ref. 25]

(c) Generic value based on a mean plus one standard deviation analysis of the high copper A508, Class 2 forging or A533, Grade B, Class 1, plate materials contained in Table G.2 of ORNL/TM-2006/530 [Ref. 26].

(d) Generic value was defined as the mean plus one standard deviation of available data from all V.C. Summer weld heats, consistent with Regulatory Guide 1.99, Revision 2 [Ref. 22].

A.5 References

1. PWR Owners Group Letter OG-21-94, Revision 0, "Electronic Endorsement for V.C. Summer Unit 1 to Participate in PA-MSC-1367R0 'Document/Reconcile/Define Basis for Reactor Vessel Material Initial RT_{NDT} and USE Values' Cafeteria Tasks 1-3," dated April 29, 2021.
2. ASME Boiler and Pressure Vessel (B&PV) Code, Section III, Division 1, Subarticle NB-2300, "Fracture Toughness Requirements for Material."
3. NUREG-0800, Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants, Chapter 5 of LWR Edition, Branch Technical Position 5-3, "Fracture Toughness Requirements," Revision 4, U.S. Nuclear Regulatory Commission, March 2019. *[Agencywide Documents Access and Management System (ADAMS) Accession Number ML18338A516]*
4. ASTM E185-82, "Standard Practice for Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels," ASTM, July 1982.
5. CMTR-RV-CGE, Revision 0, "Reactor Vessel Certified Material Test Reports for CGE [V.C. Summer Unit 1]."
6. Reactor Vessel Manual RVM-CGE, Revision 1, "V.C. Summer Unit 1 Reactor Vessel Instruction Manual," March 2017.
7. Westinghouse Report WCAP-9234, Revision 0, "South Carolina Electric and Gas Company Virgil C. Summer Nuclear Plant Unit No. 1 Reactor Vessel Radiation Surveillance Program," January 1978.
8. V.C. Summer Unit 1, Reactor Vessel Material Property data file. *[Attached in PRIME]*
9. CMTR-RV-CQL, "Reactor Vessel Certified Material Test Reports for CQL [Shearon Harris Unit 1]."
10. Westinghouse Report WCAP-10502, Revision 0, "Carolina Power and Light Company Shearon Harris Unit No. 1 Reactor Vessel Radiation Surveillance Program," May 1984.
11. CB&I Report DDP-1595, "Report in Compliance with the NRC Bulletins 78-12 & 78-12a," April 1979.
12. V.C. Summer Final Safety Analysis Report, May 2018.
13. Westinghouse Report WCAP-16305-NP, Revision 0, "V.C. Summer Heatup and Cooldown Limit Curves for Normal Operation," August 2004.
14. Westinghouse Report WCAP-16306-NP, Revision 0, "Evaluation of Pressurized Thermal Shock for V.C. Summer," August 2004.
15. Code of Federal Regulations 10 CFR 50.61, "Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events," U.S. Nuclear Regulatory Commission, Federal Register, January 4, 2010.
16. NRC Regulatory Issue Summary (RIS) 2014-11, "Information on Licensing Applications for Fracture Toughness Requirements for Ferritic Reactor Coolant Pressure Boundary Components," U.S. Nuclear Regulatory Commission, October 2014. *[ADAMS Accession Number ML14149A165]*
17. Code of Federal Regulations 10 CFR 50, Appendix G, "Fracture Toughness Requirements," U.S. Nuclear Regulatory Commission, Federal Register, December 12, 2013.

18. ASTM E185-16, "Standard Practice for Design of Surveillance Programs for Light-Water Moderated Nuclear Power Reactor Vessels," ASTM International, December 2016.

19. SCE&G Letter RC03-0112, "Virgil C. Summer Nuclear Station Docket No. 50/395 Operating License No. NPF-12 Responses to Request for Additional Information (RAI) for the Review of the License Renewal Application for Virgil C Summer Nuclear Station," date June 12, 2003. *[ADAMS Access Number ML031681125]*

20. NRC Safety Evaluation Report (SER), "Safety Evaluation Report, Related to the License Renewal of the Virgil C. Summer Nuclear Station," January 2004. *[ADAMS Access Number ML040300170, ML040300174, & ML040300177]*

21. Westinghouse Report WCAP-16298-NP, Revision 0, "Analysis of Capsule Z from the South Carolina Electric & Gas Company V. C. Summer Reactor Vessel Radiation Surveillance Program," August 2004.

22. U.S. Nuclear Regulatory Commission, Office of Nuclear Regulatory Research, Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials," May 1988. *[ADAMS Accession Number ML003740284]*

23. Framatome Report ANP-3798NP, "Analysis of Capsule Z Duke Energy Shearon Harris Nuclear Power Plant," September 2019. *[ADAMS Access Number ML19296C841]*

24. Westinghouse Report WCAP-14040-A, Revision 4, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves," May 2004. *[ADAMS Accession Number ML050120209]*

25. PWROG Report PWROG-15109-NP-A, Revision 0, "PWR Pressure Vessel Nozzle Appendix G Evaluation," January 2020. *[ADAMS Accession Number ML20024E573]*

26. Oak Ridge National Laboratory Report, ORNL/TM-2006/530, "A Physically Based Correlation of Irradiation-Induced Transition Temperature Shifts for RPV Steels," November 2007.

Attachment B
PA-MSC-1367, Tasks 1 – 3 Evaluations for V.C. Summer Unit 1 Plate and Forging Materials

B.1 V.C. Summer Unit 1 Replacement Closure Head F14362-010, Heat # 2B145585 & 2B145586

Tables B.1-1 and B.1-2 summarize all available Charpy V-notch test data and drop-weight test data taken from the V.C. Summer Unit 1 CMTRs for the Replacement Closure Head.

Table B.1-1 Charpy V-Notch Test Data for the Replacement Closure Head F14362-010

Tangential				Axial			
Temp. (°F)	**CVN Impact Energy (ft-lb)**	**Lateral Expansion (mils)**	**Shear (%)**	**Temp. (°F)**	**CVN Impact Energy (ft-lb)**	**Lateral Expansion (mils)**	**Shear (%)**
26	175	90	81	26	190	90	100
26	171	79	77	26	205	95	100
26	130	80	57	26	172	82	81
26	204	86	100	26	171	78	77
26	220	85	100	26	188	85	81
26	219	90	100	26	215	90	100

Table B.1-2 Drop-Weight Test Data for Replacement Closure Head F14362-010

Test Temperature (°F)	Drop-Weights[(a)]	T_{NDT} (°F)
-24	2–NF	-34
-34	2–F	

Note for Table B.1-2:

(a) NF = "No Fail," F = "Fail"

B.1.1 Determination of the Initial RT_{NDT}

Using the data summarized in Tables B.1-1 and B.1-2, the initial RT_{NDT} value can be determined in accordance with the ASME Code Section III, Subarticle NB-2331 requirements. Following the requirements of ASME Code Section III, Subarticle NB-2331, Paragraph (a)(2), the minimum Charpy V-notch test data is first checked at a temperature not greater than the drop-weight T_{NDT} (or NDT) plus 60°F to determine if the material exhibits at least 50 ft-lb absorbed energy and 35 mils LE in the "weak" direction. Charpy V-notch tests were conducted at 26°F, T_{NDT} + 60°F (-34°F + 60°F = 26°F). The minimum Charpy V-notch test data at this temperature exhibit a minimum of 50 ft-lb absorbed energy and 35 mils lateral expansion; therefore, the Charpy V-notch tests at T_{NDT} + 60°F satisfy the criteria. Per ASME Code Section III, Subarticle NB-2331, Paragraph (a)(2), the requirements have been met such that T_{NDT} is the initial reference temperature RT_{NDT}.

Replacement Closure Head F14362-010 Initial RT_{NDT} = -34°F

B.1.2 Determination of the Initial USE

The current 10 CFR 50, Appendix G requirements specify that USE be calculated based on ASTM E185-82. Herein, USE is calculated based on an interpretation of ASTM E185-82 that is best explained by the most recent version of the ASTM E185 manual (2016 version). Using the guidelines in ASTM E185-82 and E185-16, the average of all Charpy data ≥ 95% shear is reported as the USE. In some instances, there may be data deemed 'out of family,' which are removed from the determination of the USE based on engineering judgment. However, the use of engineering judgment to remove 'out of family' data was not necessary for this material. The USE (weak direction) is displayed below; this value is the average of each of the impact energy values contained in Table B.1-1 with shear ≥ 95%.

Replacement Closure Head F14362-010 Initial USE = Average (190, 205, 215) ft-lb
= 203 ft-lb

B.1.3 Chemistry

The Cu and Ni wt. % chemical compositions of the V.C. Summer Unit 1 reactor vessel materials were defined by a review of the available original test documentation. The material's chemical properties are defined as the average of all available data. When component specific data was not available, a generic value was defined as the mean plus one standard deviation of available data from similar materials. This method is consistent with Regulatory Guide 1.99, Revision 2, which allows the mean plus one standard deviation method to be used for conservative chemistry estimates based on generic data if component specific data is not available. The chemical compositions are summarized in Table B.1-3.

Table B.1-3 Chemistry Data for Replacement Closure Head F14362-010

Copper (wt.-%)	Nickel (wt.-%)	Source
0.03	0.73	CMTR, Doosan Analysis

Therefore, the chemical content will be defined as shown below going forward:

Replacement Closure Head F14362-010 Cu Content = 0.03 wt-%

Replacement Closure Head F14362-010 Ni Content = 0.73 wt-%

B.2 V.C. Summer Unit 1 Vessel Flange 5301-V-1, Heat # 5P5343, 4P4845, & 3P4570)

Tables B.2-1 and B.2-2 summarize all available Charpy V-notch test data and drop-weight test data taken from the V.C. Summer Unit 1 CMTRs for the Vessel Flange.

Table B.2-1 Charpy V-Notch Test Data for the Vessel Flange 5301-V-1

Tangential				Axial			
Temp. (°F)	CVN Impact Energy (ft-lb)	Lateral Expansion (mils)	Shear (%)	Temp. (°F)	CVN Impact Energy (ft-lb)	Lateral Expansion (mils)	Shear (%)
-100	6	4	0	-40	13	12	5
-100	17	11	3	-40	84	68	40
-50	87	67	29	-40	30	26	20
-50	84	65	25	-30	82	72	35
-50	143	101	46	-30	22	23	20
-20	76	57	23	-30	71	62	35
-20	127	85	50	0	99	83	70
-20	100	73	51	0	108	87	80
-10	121	87	47	0	87	68	65
40	220	84	99	60	118.5	83	80
40	190	90	97	60	155	90	100
40	186	89	98	60	90	70	65
212	180	79	100	60	130	75	80
212	176	92	100	60	151	93	100
212	174	86	100	60	137	82	85
-	-	-	-	120	156	91	95
-	-	-	-	120	131	86	80
-	-	-	-	212	145	90	100
-	-	-	-	212	161	100	100
-	-	-	-	212	153	86	100
-	-	-	-	212	150	84	100
-	-	-	-	212	166	87	100
-	-	-	-	212	158	85	100
-	-	-	-	212	193	83	100

Table B.2-2 Drop-Weight Test Data for Vessel Flange 5301-V-1

Test Temperature (°F)	Drop-Weights[a]	T_{NDT} (°F)
50	1–NF	0
10	2–NF	
0	1-F	
-10	1-F	

Note for Table B.2-2:
(a) NF = "No Fail," F = "Fail".

B.2.1 Determination of the Initial RT_{NDT}

Using the data summarized in Tables B.2-1 and B.2-2, the initial RT_{NDT} value can be determined in accordance with the ASME Code Section III, Subarticle NB-2331 requirements. Following the requirements of ASME Code Section III, Subarticle NB-2331, Paragraph (a)(2), the minimum Charpy V-notch test data is first checked at a temperature not greater than the drop-weight T_{NDT} (or NDT) plus 60°F to determine if the material exhibits at least 50 ft-lb absorbed energy and 35 mils LE in the "weak" direction. Charpy V-notch tests were conducted at 60°F, T_{NDT} + 60°F (0°F + 60°F = 60°F). The minimum Charpy V-notch test data at this temperature exhibit a minimum of 50 ft-lb absorbed energy and 35 mils lateral expansion; therefore, the Charpy V-notch tests at T_{NDT} + 60°F satisfy the criteria. Per ASME Code Section III, Subarticle NB-2331, Paragraph (a)(2), the requirements have been met such that T_{NDT} is the initial reference temperature RT_{NDT}.

Vessel Flange 5301-V-1 Initial RT_{NDT} = 0°F

B.2.2 Determination of the Initial USE

The current 10 CFR 50, Appendix G requirements specify that USE be calculated based on ASTM E185-82. Herein, USE is calculated based on an interpretation of ASTM E185-82 that is best explained by the most recent version of the ASTM E185 manual (2016 version). Using the guidelines in ASTM E185-82 and E185-16, the average of all Charpy data ≥ 95% shear is reported as the USE. In some instances, there may be data deemed 'out of family,' which are removed from the determination of the USE based on engineering judgment. However, the use of engineering judgment to remove 'out of family' data was not necessary for this material. The USE (weak direction) is displayed below; this value is the average of each of the impact energy values contained in Table B.2-1 with shear ≥ 95%.

Vessel Flange 5301-V-1 Initial USE = Average (155, 151, 156, 145, 161, 153, 150, 166, 158, 193) ft-lb
= 159 ft-lb

B.2.3 Chemistry

The Cu and Ni wt. % chemical compositions of the V.C. Summer Unit 1 reactor vessel materials were defined by a review of the available original test documentation. The material's chemical properties are defined as the average of all available data. When component specific data was not available, a generic value was defined as the mean plus one standard deviation of available data from similar materials. This method is consistent with Regulatory Guide 1.99, Revision 2, which allows the mean plus one standard deviation method to be used for conservative chemistry estimates based on generic data if component specific data is not available. The chemical compositions are summarized in Table B.2-3.

Table B.2-3 Chemistry Data for Vessel Flange 5301-V-1

Copper (wt.-%)	Nickel (wt.-%)	Source
-	0.70	CMTR, US Steel, Homestead Works, Analysis
0.153		Generic value based on a mean plus one standard deviation analysis of the high copper A508, Class 2 forging materials contained in Table G.2 of ORNL/TM-2006/530.

Therefore, the chemical content will be defined as shown below going forward:

Vessel Flange 5301-V-1 Cu Content = 0.153 wt-%

Vessel Flange 5301-V-1 Ni Content = 0.70 wt-%

B.3 V.C. Summer Unit 1 Inlet Nozzle Forging 436B-1, Heat # Q2Q41W

Tables B.3-1 and B.3-2 summarize all available Charpy V-notch test data and drop-weight test data taken from the V.C. Summer Unit 1 CMTRs for Inlet Nozzle Forging 436B-1.

Table B.3-1 Charpy V-Notch Test Data for the Inlet Nozzle Forging 436B-1

Tangential				Axial			
Temp. (°F)	CVN Impact Energy (ft-lb)	Lateral Expansion (mils)	Shear (%)	Temp. (°F)	CVN Impact Energy (ft-lb)	Lateral Expansion (mils)	Shear (%)
-100	2	1	1	-50	45	30	20
-100	3	2	1	-50	34	25	10
-100	3	3	1	-50	24	18	1
-50	64	50	50	-30	64	44	20
-50	40	30	30	-30	71	51	20
-50	45	30	30	-30	41	29	10
-20	94	66	90	-10	87	58	30
-20	82	57	80	-10	82	55	30
-20	84	61	80	-10	72	53	30
10	92	64	80	30	111	69	40
10	96	66	80	30	101	67	50
10	104	72	90	30	102	68	50
10	41	32	20	30	105	64	50
10	39	30	20	30	117	70	60
10	33	26	20	30	102	66	50
40	111	72	80	120	170	91	100
40	116	76	80	120	160	92	100
40	100	68	70	120	175	85	100
40	112	76	80	212	159	86	100
40	129	80	80	212	150	81	100
40	125	76	80	212	160	88	100
212	157	82	100	212	130	85	100
212	152	60	100	212	133	83	100
212	146	66	100	212	128	76	100

Table B.3-2 Drop-Weight Test Data for Inlet Nozzle Forging 436B-1

Test Temperature (°F)	Drop-Weights[(a)]	T_{NDT} (°F)
-10	2–NF	-20
-20	2–NF, 2–F	
-30	1–F	

Note for Table B.3-2:

(a) NF = "No Fail," F = "Fail".

B.3.1 Determination of the Initial RT_{NDT}

Using the data summarized in Tables B.3-1 and B.3-2, the initial RT_{NDT} value can be determined in accordance with the ASME Code Section III, Subarticle NB-2331 requirements. Following the requirements of ASME Code Section III, Subarticle NB-2331, Paragraph (a)(2), the minimum Charpy V-notch test data is first checked at a temperature not greater than the drop-weight T_{NDT} (or NDT) plus 60°F to determine if the material exhibits at least 50 ft-lb absorbed energy and 35 mils LE in the "weak" direction. Charpy V-notch tests were conducted at 30°F, which is less than T_{NDT} + 60°F (-20°F + 60°F = 40°F). The minimum Charpy V-notch test data at this temperature exhibit a minimum of 50 ft-lb absorbed energy and 35 mils lateral expansion; therefore, the Charpy V-notch tests at T_{NDT} + 60°F satisfy the criteria. Per ASME Code Section III, Subarticle NB-2331, Paragraph (a)(2), the requirements have been met such that T_{NDT} is the initial reference temperature RT_{NDT}.

Inlet Nozzle Forging 436B-1 Initial RT_{NDT} = -20°F

B.3.2 Determination of the Initial USE

The current 10 CFR 50, Appendix G requirements specify that USE be calculated based on ASTM E185-82. Herein, USE is calculated based on an interpretation of ASTM E185-82 that is best explained by the most recent version of the ASTM E185 manual (2016 version). Using the guidelines in ASTM E185-82 and E185-16, the average of all Charpy data ≥ 95% shear is reported as the USE. In some instances, there may be data deemed 'out of family,' which are removed from the determination of the USE based on engineering judgment. However, the use of engineering judgment to remove 'out of family' data was not necessary for this material. The Axial (weak direction) USE is displayed below; this value is the average of each of the impact energy values contained in Table B.3-1 with shear ≥ 95%.

Inlet Nozzle Forging 436B-1 Initial USE = Average (170, 160, 175, 159, 150, 160, 130, 133, 128) ft-lb
= 152 ft-lb

B.3.3 Chemistry

The Cu and Ni wt. % chemical compositions of the V.C. Summer Unit 1 reactor vessel materials were defined by a review of the available original test documentation. The material's chemical properties are defined as the average of all available data. When component specific data was not available, a generic value was defined as the mean plus one standard deviation of available data from similar materials. This method is consistent with Regulatory Guide 1.99, Revision 2, which allows the mean plus one standard deviation method to be used for conservative chemistry estimates based on generic data if component specific data is not available. The chemical compositions are summarized in Table B.3-3.

Table B.3-3 Chemistry Data for Inlet Nozzle Forging 436B-1

Copper (wt.-%)	Nickel (wt.-%)	Source
-	0.76	CMTR, Lenape Forge Analysis
-	0.76	CMTR, Verifying Analysis (performed by Bethlehem Steel)
0.127	-	Generic value for SA-508 Class 2 nozzle forgings from PWROG-15109-NP-A

Therefore, the chemical content will be defined as shown below going forward:

Inlet Nozzle Forging 436B-1 Cu Content = 0.127 wt-%

Inlet Nozzle Forging 436B-1 Ni Content = 0.76 wt-%

B.4 V.C. Summer Unit 1 Inlet Nozzle Forging 436B-2, Heat # Q2Q39W

Tables B.4-1 and B.4-2 summarize all available Charpy V-notch test data and drop-weight test data taken from the V.C. Summer Unit 1 CMTRs for Inlet Nozzle Forging 436B-2.

Table B.4-1 Charpy V-Notch Test Data for the Inlet Nozzle Forging 436B-2

Tangential				Axial			
Temp. (°F)	CVN Impact Energy (ft-lb)	Lateral Expansion (mils)	Shear (%)	Temp. (°F)	CVN Impact Energy (ft-lb)	Lateral Expansion (mils)	Shear (%)
-100	13	29	1	-50	36	26	10
-100	9	9	1	-50	14	10	1
-100	10	9	1	-50	11	8	1
-60	44	31	30	-20	82	60	40
-60	40	29	30	-20	74	55	30
-60	54	39	30	-20	63	45	20
-30	48	35	40	0	81	59	30
-30	45	35	30	0	89	63	40
-30	29	21	30	0	70	51	20
10	74	65	70	60	118	79	60
10	67	51	70	60	121	81	80
10	70	49	70	60	119	84	60
10	41	33	20	60	110	78	50
10	33	27	20	60	109	74	50
10	32	22	20	60	124	82	60
40	93	67	50	120	145	85	100
40	99	68	50	120	148	70	100
40	89	64	50	120	154	66	100
40	71	58	40	212	146	88	100
40	70	56	40	212	139	90	100
40	72	58	40	212	148	98	100
212	112	79	100	212	156	113	100
212	118	78	100	212	150	91	100
212	114	81	100	212	150	92	100

Table B.4-2 Drop-Weight Test Data for Inlet Nozzle Forging 436B-2

Test Temperature (°F)	Drop-Weights[a]	T_{NDT} (°F)
10	2–NF	0
0	1–F	
-10	1–F	
-20	1–F	
-30	1–F	

Note for Table B.4-2:
(a) NF = "No Fail," F = "Fail".

B.4.1 Determination of the Initial RT_{NDT}

Using the data summarized in Tables B.4-1 and B.4-2, the initial RT_{NDT} value can be determined in accordance with the ASME Code Section III, Subarticle NB-2331 requirements. Following the requirements of ASME Code Section III, Subarticle NB-2331, Paragraph (a)(2), the minimum Charpy V-notch test data is first checked at a temperature not greater than the drop-weight T_{NDT} (or NDT) plus 60°F to determine if the material exhibits at least 50 ft-lb absorbed energy and 35 mils LE in the "weak" direction. Charpy V-notch tests were conducted at 60°F, T_{NDT} + 60°F (0°F + 60°F = 60°F). The minimum Charpy V-notch test data at this temperature exhibit a minimum of 50 ft-lb absorbed energy and 35 mils lateral expansion; therefore, the Charpy V-notch tests at T_{NDT} + 60°F satisfy the criteria. Per ASME Code Section III, Subarticle NB-2331, Paragraph (a)(2), the requirements have been met such that T_{NDT} is the initial reference temperature RT_{NDT}.

While the tangential direction is usually considered the "strong" direction, the tangential Charpy data exhibit lower impact energies than the axial data. However, the 50 ft-lb and 35 mils LE criteria as still satisfied using the tangential impact energy data at 40°F, which is less than T_{NDT} + 60°F. Therefore, T_{NDT} still defines the initial RT_{NDT}.

Inlet Nozzle Forging 436B-2 Initial RT_{NDT} = 0°F

B.4.2 Determination of the Initial USE

The current 10 CFR 50, Appendix G requirements specify that USE be calculated based on ASTM E185-82. Herein, USE is calculated based on an interpretation of ASTM E185-82 that is best explained by the most recent version of the ASTM E185 manual (2016 version). Using the guidelines in ASTM E185-82 and E185-16, the average of all Charpy data ≥ 95% shear is reported as the USE. In some instances, there may be data deemed 'out of family,' which are removed from the determination of the USE based on engineering judgment. However, the use of engineering judgment to remove 'out of family' data was not necessary for this material. The USE (weak direction) is displayed below; this value is the average of each of the impact energy values contained in Table B.4-1 with shear ≥ 95%.

Inlet Nozzle Forging 436B-2 Initial USE = Average (145, 148, 154, 146, 139, 148, 156, 150, 150) ft-lb
= 148 ft-lb

However, while the tangential direction is usually considered the "strong" direction, the tangential Charpy data in Table B.4-1 exhibit lower impact energies at ≥ 95% shear than the axial data. Therefore, the USE will be recalculated using the tangential impact energy data with shear ≥ 95% because it represents the lower bound USE. It is noted that the BTP 5-3 methodology is NOT being implemented here, which reduces the tangentially oriented impact energies to 65% of the reported values in order to conservatively estimate axially oriented specimens. This is because axial data is available and does not need to be estimated.

Inlet Nozzle Forging 436B-2 Initial USE = Average (112, 118, 114) ft-lb
= 115 ft-lb

B.4.3 Chemistry

The Cu and Ni wt. % chemical compositions of the V.C. Summer Unit 1 reactor vessel materials were defined by a review of the available original test documentation. The material's chemical properties are defined as the average of all available data. When component specific data was not available, a generic value was defined as the mean plus one standard deviation of available data from similar materials. This method is consistent with Regulatory Guide 1.99, Revision 2, which allows the mean plus one standard deviation method to be used for conservative chemistry estimates based on generic data if component specific data is not available. The chemical compositions are summarized in Table B.4-3.

Table B.4-3 Chemistry Data for Inlet Nozzle Forging 436B-2

Copper (wt.-%)	Nickel (wt.-%)	Source
-	0.81	CMTR, Lenape Forge Analysis
-	0.82	CMTR, Verifying Analysis (performed by Bethlehem Steel)
0.127	-	Generic value for SA-508 Class 2 nozzle forgings from PWROG-15109-NP-A

Therefore, the chemical content will be defined as shown below going forward:

Inlet Nozzle Forging 436B-2 Cu Content = 0.127 wt-%

Inlet Nozzle Forging 436B-2 Ni Content = 0.82 wt-%

B.5 V.C. Summer Unit 1 Inlet Nozzle Forging 436B-3, Heat # Q2Q39W

Tables B.5-1 and B.5-2 summarize all available Charpy V-notch test data and drop-weight test data taken from the V.C. Summer Unit 1 CMTRs for Inlet Nozzle Forging 436B-3.

Table B.5-1 Charpy V-Notch Test Data for the Inlet Nozzle Forging 436B-3

Tangential				Axial			
Temp. (°F)	CVN Impact Energy (ft-lb)	Lateral Expansion (mils)	Shear (%)	Temp. (°F)	CVN Impact Energy (ft-lb)	Lateral Expansion (mils)	Shear (%)
-100	3	2	1	-50	35	25	10
-100	4	3	1	-50	40	30	10
-100	3	2	1	-50	17	12	10
-60	11	10	10	-30	59	44	20
-60	29	23	20	-30	65	48	30
-60	21	18	10	-30	45	34	20
-30	79	57	60	-20	61	46	20
-30	52	40	30	-20	50	36	20
-30	76	56	60	-20	70	51	30
10	87	60	70	40	97	61	60
10	78	55	60	40	114	70	80
10	87	62	70	40	97	65	50
10	26	20	10	40	112	74	50
10	40	31	30	40	101	64	40
10	45	35	20	40	96	60	40
40	113	69	80	120	128	75	99
40	111	74	80	120	125	81	90
40	120	76	80	120	131	77	99
40	116	75	80	212	140	82	100
40	106	69	80	212	135	80	100
40	88	64	60	212	131	88	100
212	150	63	100	212	140	89	100
212	151	66	100	212	142	66	100
212	153	51	100	212	153	73	100

Table B.5-2 Drop-Weight Test Data for Inlet Nozzle Forging 436B-3

Test Temperature (°F)	Drop-Weights[(a)]	T_{NDT} (°F)
-10	2–NF	-20
-20	1–F	
-30	1–F	

Note for Table B.5-2:

(a) NF = "No Fail," F = "Fail".

B.5.1 Determination of the Initial RT_{NDT}

Using the data summarized in Tables B.5-1 and B.5-2, the initial RT_{NDT} value can be determined in accordance with the ASME Code Section III, Subarticle NB-2331 requirements. Following the requirements of ASME Code Section III, Subarticle NB-2331, Paragraph (a)(2), the minimum Charpy V-notch test data is first checked at a temperature not greater than the drop-weight T_{NDT} (or NDT) plus 60°F to determine if the material exhibits at least 50 ft-lb absorbed energy and 35 mils LE in the "weak" direction. Charpy V-notch tests were conducted at 40°F, T_{NDT} + 60°F (-20°F + 60°F = 40°F). The minimum Charpy V-notch test data at this temperature exhibit a minimum of 50 ft-lb absorbed energy and 35 mils lateral expansion; therefore, the Charpy V-notch tests at T_{NDT} + 60°F satisfy the criteria. Per ASME Code Section III, Subarticle NB-2331, Paragraph (a)(2), the requirements have been met such that T_{NDT} is the initial reference temperature RT_{NDT}.

Inlet Nozzle Forging 436B-3 Initial RT_{NDT} = -20°F

B.5.2 Determination of the Initial USE

The current 10 CFR 50, Appendix G requirements specify that USE be calculated based on ASTM E185-82. Herein, USE is calculated based on an interpretation of ASTM E185-82 that is best explained by the most recent version of the ASTM E185 manual (2016 version). Using the guidelines in ASTM E185-82 and E185-16, the average of all Charpy data ≥ 95% shear is reported as the USE. In some instances, there may be data deemed 'out of family,' which are removed from the determination of the USE based on engineering judgment. However, the use of engineering judgment to remove 'out of family' data was not necessary for this material. The USE (weak direction) is displayed below; this value is the average of each of the impact energy values contained in Table B.5-1 with shear ≥ 95%.

Inlet Nozzle Forging 436B-3 Initial USE = Average (128, 131, 140, 135, 131, 140, 142, 153) ft-lb
= 138 ft-lb

B.5.3 Chemistry

The Cu and Ni wt. % chemical compositions of the V.C. Summer Unit 1 reactor vessel materials were defined by a review of the available original test documentation. The material's chemical properties are defined as the average of all available data. When component specific data was not available, a generic value was defined as the mean plus one standard deviation of available data from similar materials. This method is consistent with Regulatory Guide 1.99, Revision 2, which allows the mean plus one standard deviation method to be used for conservative chemistry estimates based on generic data if component specific data is not available. The chemical compositions are summarized in Table B.5-3.

Table B.5-3 Chemistry Data for Inlet Nozzle Forging 436B-3

Copper (wt.-%)	Nickel (wt.-%)	Source
-	0.81	CMTR, Lenape Forge Analysis
-	0.82	CMTR, Verifying Analysis (performed by Bethlehem Steel)
0.127	-	Generic value for SA-508 Class 2 nozzle forgings from PWROG-15109-NP-A

Therefore, the chemical content will be defined as shown below going forward:

Inlet Nozzle Forging 436B-3 Cu Content = 0.127 wt-%

Inlet Nozzle Forging 436B-3 Ni Content = 0.82 wt-%

B.6 V.C. Summer Unit 1 Outlet Nozzle Forging 437B-1, Heat # Q2Q40

Tables B.6-1 and B.6-2 summarize all available Charpy V-notch test data and drop-weight test data taken from the V.C. Summer Unit 1 CMTRs for Outlet Nozzle Forging 437B-1.

Table B.6-1 Charpy V-Notch Test Data for the Outlet Nozzle Forging 437B-1

Tangential				Axial			
Temp. (°F)	CVN Impact Energy (ft-lb)	Lateral Expansion (mils)	Shear (%)	Temp. (°F)	CVN Impact Energy (ft-lb)	Lateral Expansion (mils)	Shear (%)
-100	5	3	1	-60	80	55	30
-100	17	11	1	-60	63	48	20
-100	19	14	1	-60	69	47	20
-60	91	52	50	-30	83	63	50
-60	35	27	30	-30	64	45	30
-60	51	40	40	-30	79	67	40
-30	95	69	80	-10	86	58	50
-30	83	61	70	-10	86	60	40
-30	95	69	80	-10	64	43	30
10	110	74	50	50	155	91	100
10	94	68	40	50	138	71	85
10	112	72	50	50	128	71	75
10	33	23	10	50	127	71	75
10	74	51	60	50	134	56	85
10	87	60	70	50	135	72	75
40	95	64	50	120	155	81	100
40	64	45	40	120	138	83	100
40	86	61	50	120	128	85	100
40	113	78	70	212	147	69	100
40	99	73	60	212	158	63	100
40	125	79	70	212	144	84	100
212	174	66	100	212	176	76	100
212	175	69	100	212	190	79	100
212	177	71	100	212	201	67	100

Table B.6-2 Drop-Weight Test Data for Outlet Nozzle Forging 437B-1

Test Temperature (°F)	Drop-Weights[a]	T_{NDT} (°F)
10	2–NF	-10[b]
0	2–NF	

Notes for Table B.6-2:

(a) NF = "No Fail," F = "Fail".

(b) Drop-weight testing had no breaks at the lowest test temperature, i.e., 0°F; therefore, the NDT ≤ the next test temperature, i.e., -10°F.

B.6.1 Determination of the Initial RT_{NDT}

Using the data summarized in Tables B.6-1 and B.6-2, the initial RT_{NDT} value can be determined in accordance with the ASME Code Section III, Subarticle NB-2331 requirements. Following the requirements of ASME Code Section III, Subarticle NB-2331, Paragraph (a)(2), the minimum Charpy V-notch test data is first checked at a temperature not greater than the drop-weight TNDT (or NDT) plus 60°F to determine if the material exhibits at least 50 ft-lb absorbed energy and 35 mils LE in the "weak" direction. Charpy V-notch tests were conducted at 50°F, T_{NDT} + 60°F (-10°F + 60°F = 50°F). The minimum Charpy V-notch test data at this temperature exhibit a minimum of 50 ft-lb absorbed energy and 35 mils lateral expansion; therefore, the Charpy V-notch tests at T_{NDT} + 60°F satisfy the criteria. Per ASME Code Section III, Subarticle NB-2331, Paragraph (a)(2), the requirements have been met such that T_{NDT} is the initial reference temperature RT_{NDT}.

While the tangential direction is usually considered the "strong" direction, the tangential Charpy data exhibit lower impact energies than the axial data. However, the 50 ft-lb and 35 mils LE criteria as still satisfied using the tangential impact energy data at 40°F, which is less than T_{NDT} + 60°F. Therefore, T_{NDT} still defines the initial RT_{NDT}.

Outlet Nozzle Forging 437B-1 Initial RT_{NDT} = -10°F

B.6.2 Determination of the Initial USE

The current 10 CFR 50, Appendix G requirements specify that USE be calculated based on ASTM E185-82. Herein, USE is calculated based on an interpretation of ASTM E185-82 that is best explained by the most recent version of the ASTM E185 manual (2016 version). Using the guidelines in ASTM E185-82 and E185-16, the average of all Charpy data ≥ 95% shear is reported as the USE. In some instances, there may be data deemed 'out of family,' which are removed from the determination of the USE based on engineering judgment. However, the use of engineering judgment to remove 'out of family' data was not necessary for this material. The Axial (weak direction) USE is displayed below; this value is the average of each of the impact energy values contained in Table B.6-1 with shear ≥ 95%.

Outlet Nozzle Forging 437B-1 Initial USE = Average (155, 155, 138, 128, 147, 158, 144, 176, 190, 201) ft-lb = 159 ft-lb

B.6.3 Chemistry

The Cu and Ni wt. % chemical compositions of the V.C. Summer Unit 1 reactor vessel materials were defined by a review of the available original test documentation. The material's chemical properties are defined as the average of all available data. When component specific data was not available, a generic value was defined as the mean plus one standard deviation of available data from similar materials. This method is consistent with Regulatory Guide 1.99, Revision 2, which allows the mean plus one standard deviation method to be used for conservative chemistry estimates based on generic data if component specific data is not available. The chemical compositions are summarized in Table B.6-3.

Table B.6-3 Chemistry Data for Outlet Nozzle Forging 437B-1

Copper (wt.-%)	Nickel (wt.-%)	Source
-	0.81	CMTR, Lenape Forge Analysis
-	0.89	CMTR, Verifying Analysis (performed by Bethlehem Steel)
0.127	-	Generic value for SA-508 Class 2 nozzle forgings from PWROG-15109-NP-A

Therefore, the chemical content will be defined as shown below going forward:

Outlet Nozzle Forging 437B-1 Cu Content = 0.127 wt-%

Outlet Nozzle Forging 437B-1 Ni Content = 0.85 wt-%

B.7 V.C. Summer Unit 1 Outlet Nozzle Forging 437B-2, Heat # Q2Q40W

Tables B.7-1 and B.7-2 summarize all available Charpy V-notch test data and drop-weight test data taken from the V.C. Summer Unit 1 CMTRs for Outlet Nozzle Forging 437B-2.

Table B.7-1 Charpy V-Notch Test Data for the Outlet Nozzle Forging 437B-2

Tangential				Axial			
Temp. (°F)	CVN Impact Energy (ft-lb)	Lateral Expansion (mils)	Shear (%)	Temp. (°F)	CVN Impact Energy (ft-lb)	Lateral Expansion (mils)	Shear (%)
-100	5	3	1	-60	63	45	20
-100	29	18	1	-60	44	30	10
-100	28	19	1	-60	65	47	20
-60	64	50	60	-40	46	35	20
-60	74	58	70	-40	77	54	30
-60	15	10	10	-40	79	54	30
-30	66	40	40	-10	76	54	30
-30	101	72	80	-10	74	55	40
-30	40	30	20	-10	97	67	50
10	176	87	99	50	155	83	100
10	106	76	80	50	159	84	100
10	112	74	80	50	168	80	100
10	71	50	70	50	138	71	75
10	72	51	70	50	168	90	100
10	56	40	90	50	129	71	70
40	128	79	70	120	152	88	100
40	111	75	70	120	159	85	100
40	158	84	80	120	169	84	100
40	128	79	70	212	165	87	100
40	111	75	70	212	170	74	100
40	158	84	90	212	161	89	100
212	168	64	100	212	170	89	100
212	161	66	100	212	190	81	100
212	172	84	100	212	157	94	100

Table B.7-2 Drop-Weight Test Data for Outlet Nozzle Forging 437B-2

Test Temperature (°F)	Drop-Weights[a]	T_{NDT} (°F)
0	2–NF	-10
-10	1–F	
-20	1–F	
-30	1–F	

Note for Table B.7-2:

(a) NF = "No Fail," F = "Fail".

B.7.1 Determination of the Initial RT_{NDT}

Using the data summarized in Tables B.7-1 and B.7-2, the initial RT_{NDT} value can be determined in accordance with the ASME Code Section III, Subarticle NB-2331 requirements. Following the requirements of ASME Code Section III, Subarticle NB-2331, Paragraph (a)(2), the minimum Charpy V-notch test data is first checked at a temperature not greater than the drop-weight T_{NDT} (or NDT) plus 60°F to determine if the material exhibits at least 50 ft-lb absorbed energy and 35 mils LE in the "weak" direction. Charpy V-notch tests were conducted at 50°F, T_{NDT} + 60°F (-10°F + 60°F = 50°F). The minimum Charpy V-notch test data at this temperature exhibit a minimum of 50 ft-lb absorbed energy and 35 mils lateral expansion; therefore, the Charpy V-notch tests at T_{NDT} + 60°F satisfy the criteria. Per ASME Code Section III, Subarticle NB-2331, Paragraph (a)(2), the requirements have been met such that T_{NDT} is the initial reference temperature RT_{NDT}.

While the tangential direction is usually considered the "strong" direction, the tangential Charpy data exhibit lower impact energies than the axial data. However, the 50 ft-lb and 35 mils LE criteria as still satisfied using the tangential impact energy data at 40°F, which is less than T_{NDT} + 60°F. Therefore, T_{NDT} still defines the initial RT_{NDT}.

Outlet Nozzle Forging 437B-2 Initial RT_{NDT} = -10°F

B.7.2 Determination of the Initial USE

The current 10 CFR 50, Appendix G requirements specify that USE be calculated based on ASTM E185-82. Herein, USE is calculated based on an interpretation of ASTM E185-82 that is best explained by the most recent version of the ASTM E185 manual (2016 version). Using the guidelines in ASTM E185-82 and E185-16, the average of all Charpy data ≥ 95% shear is reported as the USE. In some instances, there may be data deemed 'out of family,' which are removed from the determination of the USE based on engineering judgment. However, the use of engineering judgment to remove 'out of family' data was not necessary for this material. The Axial (weak direction) USE is displayed below; this value is the average of each of the impact energy values contained in Table B.7-1 with shear ≥ 95%.

Outlet Nozzle Forging 437B-2 Initial USE = Average (155, 159, 168, 168, 152, 159, 169, 165, 170, 161, 170, 190, 157) ft-lb
= 165 ft-lb

B.7.3 Chemistry

The Cu and Ni wt. % chemical compositions of the V.C. Summer Unit 1 reactor vessel materials were defined by a review of the available original test documentation. The material's chemical properties are defined as the average of all available data. When component specific data was not available, a generic value was defined as the mean plus one standard deviation of available data from similar materials. This method is consistent with Regulatory Guide 1.99, Revision 2, which allows the mean plus one standard deviation method to be used for conservative chemistry estimates based on generic data if component specific data is not available. The chemical compositions are summarized in Table B.7-3.

Table B.7-3 Chemistry Data for Outlet Nozzle Forging 437B-2

Copper (wt.-%)	Nickel (wt.-%)	Source
-	0.80	CMTR, Lenape Forge Analysis
0.127	-	Generic value for SA-508 Class 2 nozzle forgings from PWROG-15109-NP-A

Therefore, the chemical content will be defined as shown below going forward:

Outlet Nozzle Forging 437B-2 Cu Content = 0.127 wt-%

Outlet Nozzle Forging 437B-2 Ni Content = 0.80 wt-%

B.8 V.C. Summer Unit 1 Outlet Nozzle Forging 437B-3, Heat # Q2Q44W

Tables B.8-1 and B.8-2 summarize all available Charpy V-notch test data and drop-weight test data taken from the V.C. Summer Unit 1 CMTRs for Outlet Nozzle Forging 437B-3.

Table B.8-1 Charpy V-Notch Test Data for the Outlet Nozzle Forging 437B-3

Tangential				Axial			
Temp. (°F)	CVN Impact Energy (ft-lb)	Lateral Expansion (mils)	Shear (%)	Temp. (°F)	CVN Impact Energy (ft-lb)	Lateral Expansion (mils)	Shear (%)
-100	5	2	1	-50	21	14	1
-100	8	6	1	-50	33	23	10
-100	4	1	1	-50	87	61	40
-60	33	24	10	-40	54	38	20
-60	27	21	20	-40	8	5	1
-60	80	57	50	-40	15	11	1
-30	85	64	90	-20	88	60	50
-30	43	31	30	-20	67	50	40
-30	71	54	90	-20	96	67	50
10	104	73	80	40	121	77	65
10	117	78	80	40	119	71	70
10	103	71	80	40	123	79	65
10	76	55	40	40	130	80	75
10	37	26	20	40	130	75	75
10	75	54	50	40	119	68	70
40	116	76	70	120	126	70	90
40	111	73	70	120	129	69	80
40	115	74	70	120	149	85	100
40	127	79	80	212	150	62	100
40	121	78	80	212	144	81	100
40	120	73	70	212	156	82	100
212	162	66	100	212	160	85	100
212	164	70	100	212	160	81	100
212	163	86	100	212	168	79	100

Table B.8-2 Drop-Weight Test Data for Outlet Nozzle Forging 437B-3

Test Temperature (°F)	Drop-Weights[a]	T_{NDT} (°F)
20	2–NF	0[b]
0	1–NF, 1–F	
-10	2–NF	
-20	1–F	
-30	1–F	

Notes for Table B.8-2:

(a) NF = "No Fail," F = "Fail".

(b) Two test presented inconsistent results for drop-weight. The more conservative result will be used to define the material properties, i.e., 0°F.

B.8.1 Determination of the Initial RT_{NDT}

Using the data summarized in Tables B.8-1 and B.8-2, the initial RT_{NDT} value can be determined in accordance with the ASME Code Section III, Subarticle NB-2331 requirements. Following the requirements of ASME Code Section III, Subarticle NB-2331, Paragraph (a)(2), the minimum Charpy V-notch test data is first checked at a temperature not greater than the drop-weight T_{NDT} (or NDT) plus 60°F to determine if the material exhibits at least 50 ft-lb absorbed energy and 35 mils LE in the "weak" direction. Charpy V-notch tests were conducted at 40°F, which is less than T_{NDT} + 60°F (0°F + 60°F = 60°F). The minimum Charpy V-notch test data at this temperature exhibit a minimum of 50 ft-lb absorbed energy and 35 mils lateral expansion; therefore, the Charpy V-notch tests at T_{NDT} + 60°F satisfy the criteria. Per ASME Code Section III, Subarticle NB-2331, Paragraph (a)(2), the requirements have been met such that T_{NDT} is the initial reference temperature RT_{NDT}.

While the tangential direction is usually considered the "strong" direction, the tangential Charpy data exhibit lower impact energies than the axial data. However, the 50 ft-lb and 35 mils LE criteria as still satisfied using the tangential impact energy data at 40°F, which is less than T_{NDT} + 60°F. Therefore, T_{NDT} still defines the initial RT_{NDT}.

Outlet Nozzle Forging 437B-3 Initial RT_{NDT} = 0°F

B.8.2 Determination of the Initial USE

The current 10 CFR 50, Appendix G requirements specify that USE be calculated based on ASTM E185-82. Herein, USE is calculated based on an interpretation of ASTM E185-82 that is best explained by the most recent version of the ASTM E185 manual (2016 version). Using the guidelines in ASTM E185-82 and E185-16, the average of all Charpy data ≥ 95% shear is reported as the USE. In some instances, there may be data deemed 'out of family,' which are removed from the determination of the USE based on engineering judgment. However, the use of engineering judgment to remove 'out of family' data was not necessary for this material. The Axial (weak direction) USE is displayed below; this value is the average of each of the impact energy values contained in Table B.8-1 with shear ≥ 95%.

Outlet Nozzle Forging 437B-3 Initial USE = Average (149, 150, 144, 156, 160, 160, 168) ft-lb
= 155 ft-lb

B.8.3 Chemistry

The Cu and Ni wt. % chemical compositions of the V.C. Summer Unit 1 reactor vessel materials were defined by a review of the available original test documentation. The material's chemical properties are defined as the average of all available data. When component specific data was not available, a generic value was defined as the mean plus one standard deviation of available data from similar materials. This method is consistent with Regulatory Guide 1.99, Revision 2, which allows the mean plus one standard deviation method to be used for conservative chemistry estimates based on generic data if component specific data is not available. The chemical compositions are summarized in Table B.8-3.

Table B.8-3 Chemistry Data for Outlet Nozzle Forging 437B-3

Copper (wt.-%)	Nickel (wt.-%)	Source
-	0.78	CMTR, Lenape Forge Analysis
	0.77	CMTR, Verifying Analysis (performed by Bethlehem Steel)
0.127	-	Generic value for SA-508 Class 2 nozzle forgings from PWROG-15109-NP-A

Therefore, the chemical content will be defined as shown below going forward:

Outlet Nozzle Forging 437B-3 Cu Content = 0.127 wt-%

Outlet Nozzle Forging 437B-3 Ni Content = 0.78 wt-%

B.9 V.C. Summer Unit 1 Nozzle Shell Plate, Heat # C9955-2

Tables B.9-1 and B.9-2 summarize all available Charpy V-notch test data and drop-weight test data taken from the V.C. Summer Unit 1 CMTRs for Nozzle Shell Plate, Heat # C9955-2.

Table B.9-1 Charpy V-Notch Test Data for the Nozzle Shell Plate, Heat # C9955-2

Longitudinal				Transverse			
Temp. (°F)	CVN Impact Energy (ft-lb)	Lateral Expansion (mils)	Shear (%)	Temp. (°F)	CVN Impact Energy (ft-lb)	Lateral Expansion (mils)	Shear (%)
-100	4[(a)]	2[(a)]	1	-20	11[(a)]	13[(a)]	15
-100	4	3	1	-20	15.5	18	15
-100	4	2	1	-20	15	17	20
-50	15	11[(a)]	10	40	49	43	50
-50	15	15	10	40	55.5	49	45
-50	12[(a)]	14	10	40	34[(a)]	34[(a)]	35
10	55	48	50	50	59	51	55
10	66	47[(a)]	50	50	46.5	43	45
10	52[(a)]	49	50	50	38[(a)]	40[(a)]	60
40	50[(a)]	50	40	70	43[(a)]	45[(a)]	50
40	65	43	40	70	55	53	55
40	61	37[(a)]	40	70	48	45	50
100	113	70[(a)]	70	120	88	72	75
100	99[(a)]	79	70	120	85[(a)]	71[(a)]	75
100	105	82	70	120	92	72	80
212	147	95[(a)(b)]	99	212	103	78	100
212	148	96	99	212	106.5	76[(a)(b)]	100
212	140[(a)(b)]	96	99	212	93[(a)]	78	100

Notes for Table B.9-1:

(a) Minimum value used in the CVGRAPH plots in accordance with ASME Code III Subarticle NB-2331 criteria.

(b) The value fixed as the upper shelf in CVGRAPH plots.

Table B.9-2 Drop-Weight Test Data for Nozzle Shell Plate, Heat # C9955-2

Test Temperature (°F)	Drop-Weights[(a)]	T_{NDT} (°F)
0	1–NF	-20
-10	2–NF	
-20	1–NF, 1–F	
-30	1–F	

Note for Table B.9-2:
(a) NF = "No Fail," F = "Fail".

B.9.1 Determination of the Initial RT_{NDT}

Using the data summarized in Tables B.9-1 and B.9-2, the initial RT_{NDT} value can be determined in accordance with the ASME Code Section III, Subarticle NB-2331 requirements. Following the requirements of ASME Code Section III, Subarticle NB-2331, Paragraph (a)(2), the minimum Charpy V-notch test data is first checked at a temperature not greater than the drop-weight T_{NDT} (or NDT) plus 60°F to determine if the material exhibits at least 50 ft-lb absorbed energy and 35 mils LE in the "weak" direction. Charpy V-notch tests were conducted at 40°F, T_{NDT} + 60°F (-20°F + 60°F = 40°F). The minimum Charpy V-notch test data at this temperature did **NOT** exhibit a minimum of 50 ft-lb absorbed energy and 35 mils lateral expansion; therefore, the Charpy V-notch tests at T_{NDT} + 60°F would **NOT** satisfy the criteria.

To precisely determine the temperature at which 50 ft-lb and 35 mils LE were obtained on the specimens, the unirradiated Charpy V-notch data may be plotted and fit using a hyperbolic tangent curve-fitting software, CVGRAPH. Only the minimum data points at each Charpy V-notch test temperature were used as input to the curve-fitting software, in accordance with ASME Code Section III, Subarticle NB-2331, Paragraph (a)(4). When plotting, the USE is fixed to the minimum Charpy impact energy or lateral expansion used in the plot which experience ≥ 95% shear. The resulting CVGRAPH figures are contained in the following pages for Charpy V-notch absorbed energy and lateral expansion.

Using these figures, the temperature at which 50 ft-lb absorbed energy and 35 mils lateral expansion were achieved may be obtained. The absorbed energy test data is more conservative than the lateral expansion test data; therefore, it becomes the dominant data set in defining initial RT_{NDT}.

$T_{50\ ft\text{-}lb}$ = 68.9°F
$T_{35\ mils}$ = 42°F
T_{Cv} = Max [$T_{50\ ft\text{-}lb}$, $T_{35\ mil}$] = Max [68.9°F, 42°F]
T_{Cv} = 68.9°F

Following the requirements of ASME Code Section III, Subarticle NB-2331, Paragraph (a)(3), the initial RT_{NDT} is the higher of T_{NDT} (determined from the drop-weight tests) and T_{Cv} (determined above) minus 60°F.

RT_{NDT} = Max [T_{NDT}, T_{CV} - 60°F]
RT_{NDT} = Max [-20°F, 68.9°F - 60°F] = Max [-20°F, 8.9°F]

Nozzle Shell Plate, Heat # C9955-2 Initial RT_{NDT} = 9°F

B.9.2 Determination of the Initial USE

The current 10 CFR 50, Appendix G requirements specify that USE be calculated based on ASTM E185-82. Herein, USE is calculated based on an interpretation of ASTM E185-82 that is best explained by the most recent version of the ASTM E185 manual (2016 version). Using the guidelines in ASTM E185-82 and E185-16, the average of all Charpy data ≥ 95% shear is reported as the USE. In some instances, there may be data deemed 'out of family,' which are removed from the determination of the USE based on engineering judgment. However, the use of engineering judgment to remove 'out of family' data was not necessary for this material. The Transverse (weak direction) USE is displayed below; this value is the average of each of the impact energy values contained in Table B.9-1 with shear ≥ 95%.

Nozzle Shell Plate, Heat # C9955-2 Initial USE = Average (103, 106.5, 93) ft-lb
= 101 ft-lb

B.9.3 Chemistry

The Cu and Ni wt. % chemical compositions of the V.C. Summer Unit 1 reactor vessel materials were defined by a review of the available original test documentation. The material's chemical properties are defined as the average of all available data. When component specific data was not available, a generic value was defined as the mean plus one standard deviation of available data from similar materials. This method is consistent with Regulatory Guide 1.99, Revision 2, which allows the mean plus one standard deviation method to be used for conservative chemistry estimates based on generic data if component specific data is not available. The chemical compositions are summarized in Table B.9-3.

Table B.9-3 Chemistry Data for Nozzle Shell Plate, Heat # C9955-2

Copper (wt.-%)	Nickel (wt.-%)	Source
0.13[(a)]	0.57	CMTR, Lukens Steel Analysis

Note for Table B.9-3:
(a) The chemistry value is based on a ladle analysis since no material check value is available.

Therefore, the chemical content will be defined as shown below going forward:

Nozzle Shell Plate, Heat # C9955-2 Cu Content = 0.13 wt-%

Nozzle Shell Plate, Heat # C9955-2 Ni Content = 0.57 wt-%

Figure B.9-1 Nozzle Shell Plate, Heat # C9955-2
Plot of Measured Transverse Direction CVN Data

CVGraph 6.02: Hyperbolic Tangent Curve Printed on 9/14/2021 1:16 PM

A = 47.60 B = 45.40 C = 64.65 T0 = 65.38 D = 0.00

Correlation Coefficient = 0.989

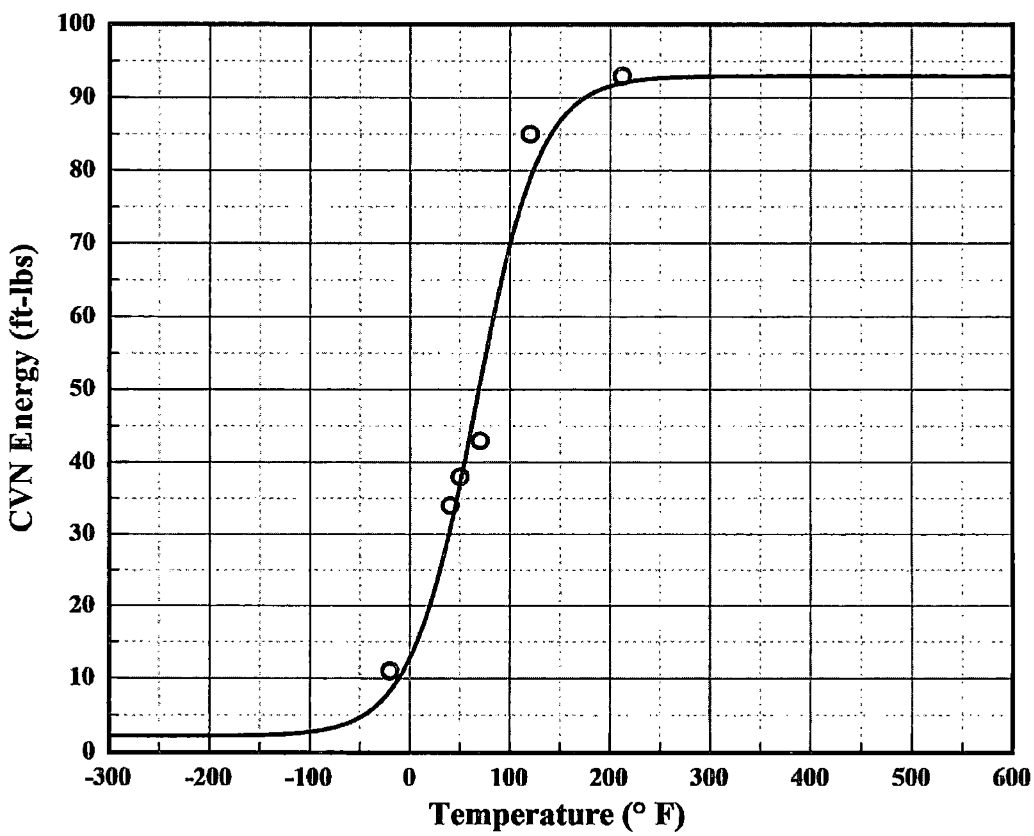
Equation is A + B * [Tanh((T-T0)/(C+DT))]

Upper Shelf Energy = 93.00 (Fixed) Lower Shelf Energy = 2.20 (Fixed)

Temp@30 ft-lbs= 39.00° F Temp@35 ft-lbs= 47.00° F Temp@50 ft-lbs= 68.90° F

Plant: **V.C. Summer** Material: **SA533B1** Heat: **C9955-2**

Orientation: **Trans.** Capsule: **Unirrad** Fluence: **0.00E+000 n/cm²**



CVGraph 6.02 09/14/2021 Page 1/2

Figure B.9-1 Nozzle Shell Plate, Heat # C9955-2
Plot of Measured Transverse Direction CVN Data (cont.)

Plant: **V.C. Summer**
Orientation: **Trans.**

Material: **SA533B1**
Capsule: **Unirrad**

Heat: **C9955-2**
Fluence: **0.00E+000 n/cm²**

Charpy V-Notch Data

Temperature (° F)	Input CVN	Computed CVN	Differential
-20	11.0	8.2	2.76
40	34.0	30.6	3.36
50	38.0	37.0	1.00
70	43.0	50.8	-7.84
120	85.0	78.9	6.15
212	93.0	92.0	0.96

CVGraph 6.02 09/14/2021 Page 2/2

Figure B.9-2 Nozzle Shell Plate, Heat # C9955-2
Plot of Measured Transverse Direction Lateral Expansion Data

CVGraph 6.02: Hyperbolic Tangent Curve Printed on 9/14/2021 1:35 PM

A = 38.50 B = 37.50 C = 73.25 T0 = 48.78 D = 0.00

Correlation Coefficient = 0.993

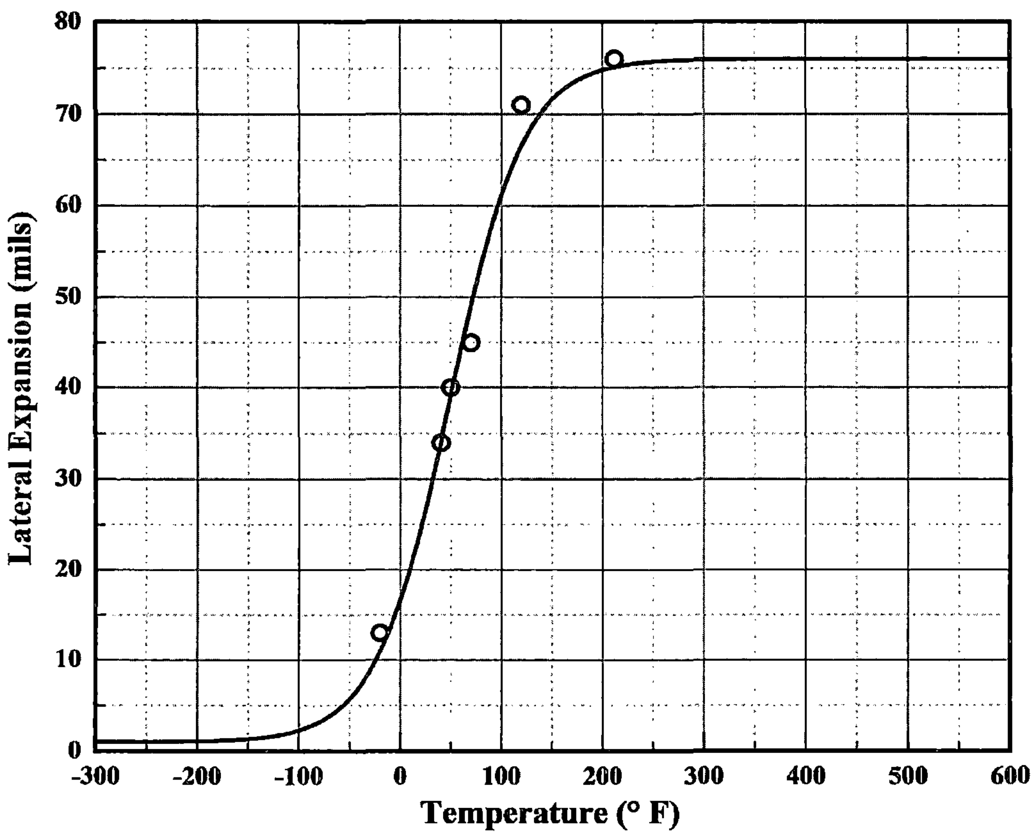
Equation is A + B * [Tanh((T-T0)/(C+DT))]

Upper Shelf L.E. = 76.00 (Fixed) Lower Shelf L.E. = 1.00 (Fixed)

Temp@35 mils= 42.00° F

Plant: **V.C. Summer** Material: **SA533B1** Heat: **C9955-2**

Orientation: **Trans.** Capsule: **Unirrad** Fluence: **0.00E+000 n/cm²**



CVGraph 6.02 09/14/2021 Page 1/2

Figure B.9-2 Nozzle Shell Plate, Heat # C9955-2
Plot of Measured Transverse Direction Lateral Expansion Data (cont.)

Plant: **V.C. Summer**
Orientation: **Trans.**

Material: **SA533B1**
Capsule: **Unirrad**

Heat: **C9955-2**
Fluence: **0.00E+000 n/cm²**

Charpy V-Notch Data

Temperature (° F)	Input L. E.	Computed L. E.	Differential
-20	13.0	10.9	2.05
40	34.0	34.0	-0.03
50	40.0	39.1	0.87
70	45.0	49.1	-4.07
120	71.0	66.6	4.39
212	76.0	75.1	0.86

CVGraph 6.02 09/14/2021 Page 2/2

B.10 V.C. Summer Unit 1 Nozzle Shell Plate, Heat # C0123-2

Tables B.10-1 and B.10-2 summarize all available Charpy V-notch test data and drop-weight test data taken from the V.C. Summer Unit 1 CMTRs for Nozzle Shell Plate, Heat # C0123-2.

Table B.10-1 Charpy V-Notch Test Data for the Nozzle Shell Plate, Heat # C0123-2

Longitudinal				Transverse			
Temp. (°F)	CVN Impact Energy (ft-lb)	Lateral Expansion (mils)	Shear (%)	Temp. (°F)	CVN Impact Energy (ft-lb)	Lateral Expansion (mils)	Shear (%)
-100	5	4	1	-30	21	19	20
-100	5	4	1	-30	16.5	16	25
-100	4[(a)]	3	1	-30	14[(a)]	14[(a)]	15
-50	13	12	10	30	34	33	40
-50	15	15	10	30	41	37	30
-50	15	15	10	30	34[(a)]	31[(a)]	35
10	38[(a)]	36	30	50	41	40	45
10	44	34	30	50	37	35	35
10	61	58	30	50	37	38	50
40	85	61	60	70	41.5[(a)]	41[(a)]	50
40	76[(a)]	68	60	70	55	52	55
40	94	60	60	70	44	43	50
100	120	87	90	120	72[(a)]	60[(a)]	60
100	119[(a)]	85	90	120	75	60	60
100	122	80	90	120	80	66	60
212	145[(a)(b)]	95	99	212	97	80	100
212	148	90[(a)(b)]	99	212	87[(a)(b)]	78	95
212	158	92	99	212	89	71[(a)(b)]	100

Notes for Table B.10-1:

(a) Minimum value used in the CVGRAPH plots in accordance with ASME Code III Subarticle NB-2331 criteria.

(b) The value fixed as the upper shelf in CVGRAPH plots.

Table B.10-2 Drop-Weight Test Data for Nozzle Shell Plate, Heat # C0123-2

Test Temperature (°F)	Drop-Weights[a]	T_{NDT} (°F)
-10	1–NF	-30
-20	2–NF	
-30	1–F	
-40	1–F	
-50	1–F	

Note for Table B.10-2:
(a) NF = "No Fail," F = "Fail".

B.10.1 Determination of the Initial RT_{NDT}

Using the data summarized in Tables B.10-1 and B.10-2, the initial RT_{NDT} value can be determined in accordance with the ASME Code Section III, Subarticle NB-2331 requirements. Following the requirements of ASME Code Section III, Subarticle NB-2331, Paragraph (a)(2), the minimum Charpy V-notch test data is first checked at a temperature not greater than the drop-weight TNDT (or NDT) plus 60°F to determine if the material exhibits at least 50 ft-lb absorbed energy and 35 mils LE in the "weak" direction. Charpy V-notch tests were conducted at 30°F, T_{NDT} + 60°F (-30°F + 60°F = 30°F). The minimum Charpy V-notch test data at this temperature did **NOT** exhibit a minimum of 50 ft-lb absorbed energy and 35 mils lateral expansion; therefore, the Charpy V-notch tests at T_{NDT} + 60°F would **NOT** satisfy the criteria.

To precisely determine the temperature at which 50 ft-lb and 35 mils LE were obtained on the specimens, the unirradiated Charpy V-notch data may be plotted and fit using a hyperbolic tangent curve-fitting software, CVGRAPH. Only the minimum data points at each Charpy V-notch test temperature were used as input to the curve-fitting software, in accordance with ASME Code Section III, Subarticle NB-2331, Paragraph (a)(4). When plotting, the USE is fixed to the minimum Charpy impact energy or lateral expansion used in the plot which experience ≥ 95% shear. The resulting CVGRAPH figures are contained in the following pages for Charpy V-notch absorbed energy and lateral expansion.

Using these figures, the temperature at which 50 ft-lb absorbed energy and 35 mils lateral expansion were achieved may be obtained. The absorbed energy test data is more conservative than the lateral expansion test data; therefore, it becomes the dominant data set in defining initial RT_{NDT}.

$T_{50\ ft\text{-}lb}$ = 74.9°F
$T_{35\ mils}$ = 45.9°F
T_{Cv} = Max [$T_{50\ ft\text{-}lb}$, $T_{35\ mil}$] = Max [74.9°F, 45.9°F]
T_{Cv} = 74.9°F

Following the requirements of ASME Code Section III, Subarticle NB-2331, Paragraph (a)(3), the initial RT_{NDT} is the higher of T_{NDT} (determined from the drop-weight tests) and T_{Cv} (determined above) minus 60°F.

RT_{NDT} = Max [T_{NDT}, T_{CV} - 60°F]
RT_{NDT} = Max [-30°F, 74.9°F - 60°F] = Max [-30°F, 14.9°F]

Nozzle Shell Plate, Heat # C0123-2 Initial RT_{NDT} = 15°F

B.10.2 Determination of the Initial USE

The current 10 CFR 50, Appendix G requirements specify that USE be calculated based on ASTM E185-82. Herein, USE is calculated based on an interpretation of ASTM E185-82 that is best explained by the most recent version of the ASTM E185 manual (2016 version). Using the guidelines in ASTM E185-82 and E185-16, the average of all Charpy data ≥ 95% shear is reported as the USE. In some instances, there may be data deemed 'out of family,' which are removed from the determination of the USE based on engineering judgment. However, the use of engineering judgment to remove 'out of family' data was not necessary for this material. The Transverse (weak direction) USE is displayed below; this value is the average of each of the impact energy values contained in Table B.10-1 with shear ≥ 95%.

Nozzle Shell Plate, Heat # C0123-2 Initial USE = Average (97, 87, 89) ft-lb
= 91 ft-lb

B.10.3 Chemistry

The Cu and Ni wt. % chemical compositions of the V.C. Summer Unit 1 reactor vessel materials were defined by a review of the available original test documentation. The material's chemical properties are defined as the average of all available data. When component specific data was not available, a generic value was defined as the mean plus one standard deviation of available data from similar materials. This method is consistent with Regulatory Guide 1.99, Revision 2, which allows the mean plus one standard deviation method to be used for conservative chemistry estimates based on generic data if component specific data is not available. The chemical compositions are summarized in Table B.10-3.

Table B.10-3 Chemistry Data for Nozzle Shell Plate, Heat # C0123-2

Copper (wt.-%)	Nickel (wt.-%)	Source
0.12[(a)]	0.58	CMTR, Lukens Steel Analysis

Note for Table B.10-3:

(a) The chemistry value is based on a ladle analysis since no material check value is available.

Therefore, the chemical content will be defined as shown below going forward:

Nozzle Shell Plate, Heat # C0123-2 Cu Content = 0.12 wt-%

Nozzle Shell Plate, Heat # C0123-2 Ni Content = 0.58 wt-%

Figure B.10-1 Nozzle Shell Plate, Heat # C0123-2
Plot of Measured Transverse Direction CVN Data

CVGraph 6.02: Hyperbolic Tangent Curve Printed on 9/14/2021 2:10 PM

A = 44.60 B = 42.40 C = 91.27 T0 = 63.19 D = 0.00

Correlation Coefficient = 0.989

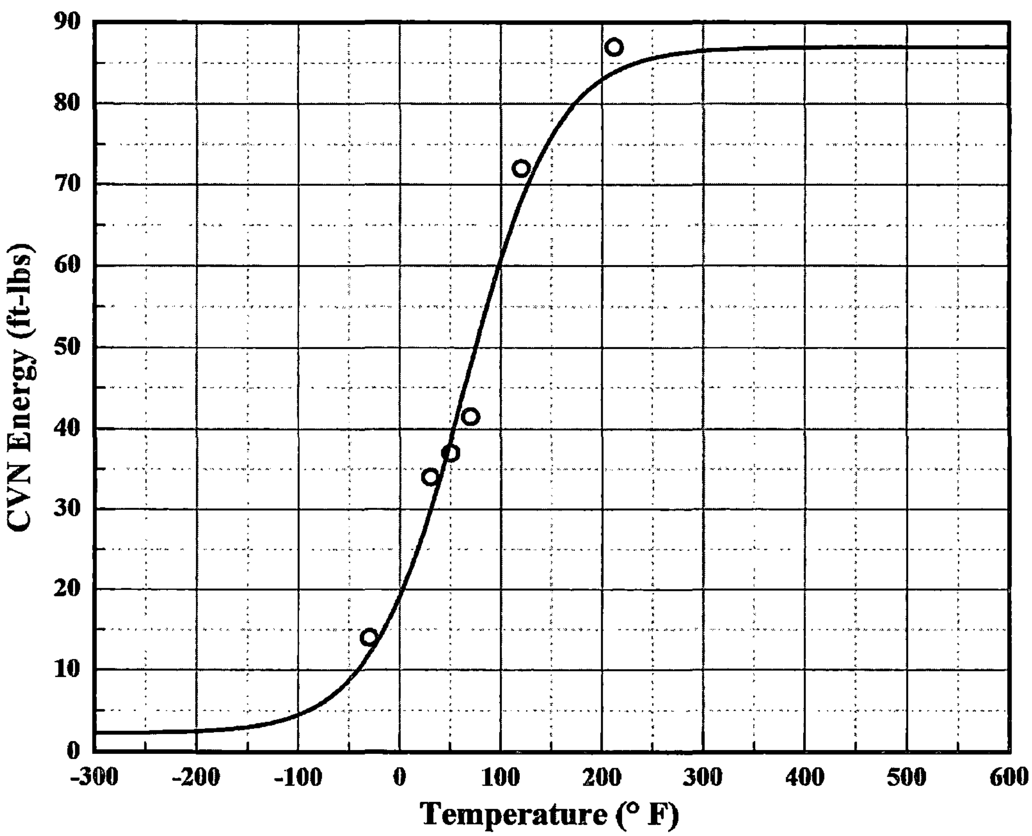
Equation is A + B * [Tanh((T-T0)/(C+DT))]

Upper Shelf Energy = 87.00 (Fixed) Lower Shelf Energy = 2.20 (Fixed)

Temp@30 ft-lbs= 30.50° F Temp@35 ft-lbs= 42.20° F Temp@50 ft-lbs= 74.90° F

Plant: **V.C. Summer** Material: **SA533B1** Heat: **C0123-2**

Orientation: **Trans.** Capsule: **unirrad** Fluence: **0.00E+000 n/cm²**



CVGraph 6.02 09/14/2021 Page 1/2

Figure B.10-1 Nozzle Shell Plate, Heat # C0123-2
Plot of Measured Transverse Direction CVN Data (cont.)

Plant: **V.C. Summer** Material: **SA533B1** Heat: **C0123-2**
Orientation: **Trans.** Capsule: **unirrad** Fluence: **0.00E+000 n/cm²**

Charpy V-Notch Data

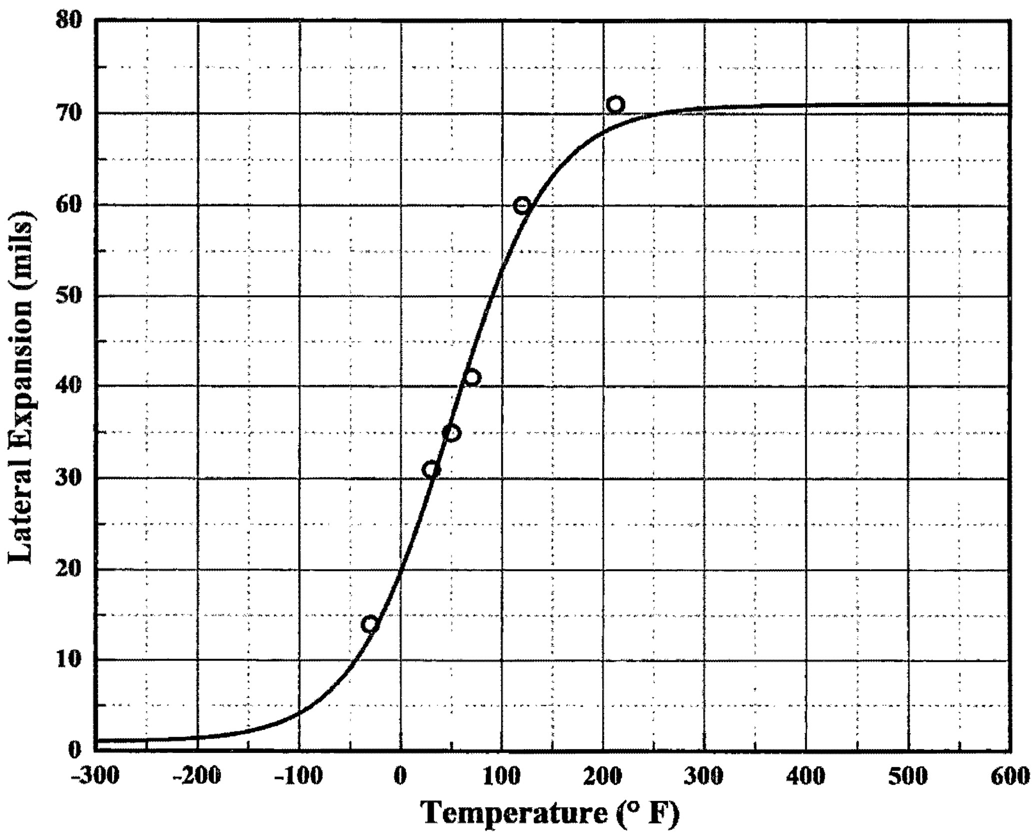
Temperature (° F)	Input CVN	Computed CVN	Differential
-30	14.0	11.9	2.06
30	34.0	29.8	4.17
50	37.0	38.5	-1.51
70	41.5	47.8	-6.26
120	72.0	68.0	3.96
212	87.0	83.9	3.13

Figure B.10-2 Nozzle Shell Plate, Heat # C0123-2
Plot of Measured Transverse Direction Lateral Expansion Data

CVGraph 6.02: Hyperbolic Tangent Curve Printed on 9/14/2021 2:12 PM
A = 36.00 B = 35.00 C = 97.41 T0 = 48.63 D = 0.00
Correlation Coefficient = 0.995
Equation is A + B * [Tanh((T-T0)/(C+DT))]

Upper Shelf L.E. = 71.00 (Fixed) Lower Shelf L.E. = 1.00 (Fixed)
Temp@35 mils= 45.90° F

Plant: **V.C. Summer** Material: **SA533B1** Heat: **C0123-2**
Orientation: **Trans.** Capsule: **unirrad** Fluence: **0.00E+000 n/cm²**



CVGraph 6.02 09/14/2021 Page 1/2

Figure B.10-2 Nozzle Shell Plate, Heat # C0123-2
Plot of Measured Transverse Direction Lateral Expansion Data (cont.)

Plant: **V.C. Summer** Material: **SA533B1** Heat: **C0123-2**
Orientation: **Trans.** Capsule: **unirrad** Fluence: **0.00E+000 n/cm²**

Charpy V-Notch Data

Temperature (° F)	Input L. E.	Computed L. E.	Differential
-30	14.0	12.6	1.38
30	31.0	29.4	1.61
50	35.0	36.5	-1.49
70	41.0	43.6	-2.56
120	60.0	57.9	2.14
212	71.0	68.6	2.36

CVGraph 6.02 09/14/2021 Page 2/2

B.11 V.C. Summer Unit 1 Intermediate Shell Plate, Heat # A9154-1

Tables B.11-1 and B.11-2 summarize all available Charpy V-notch test data and drop-weight test data taken from the V.C. Summer Unit 1 CMTRs for Intermediate Shell Plate, Heat # A9154 1. This data is combined with the additional data available from V.C. Summer Unit 1 surveillance program documented in WCAP-9134.

Table B.11-1 Charpy V-Notch Test Data for the Intermediate Shell Plate, Heat # A9154-1

Longitudinal				Transverse			
Temp. (°F)	CVN Impact Energy (ft-lb)	Lateral Expansion (mils)	Shear (%)	Temp. (°F)	CVN Impact Energy (ft-lb)	Lateral Expansion (mils)	Shear (%)
-100	5	5	1	-40	7.5(a)	2(a)	9
-100	3(a)	6	1	-40	9	4	10
-100	4	2(a)	1	-20	12.5(a)	15(a)	10
-40	7.5(a)	3(a)	8	-20	20	19	20
-40	27	16	14	-20	19	18	20
-20	17	18	20	0	29(a)	22(a)	13
-20	10(a)	16	20	0	31	26	16
-20	15	13(a)	20	40	36	34	35
-20	20	14	20	40	41	40	45
0	49.5(a)	35(a)	20	40	38	37	30
0	55	40	25	40	25	24	42
10	54	46	40	40	23(a)	22(a)	25
10	50	44	40	40	35	28	23
10	48(a)	42(a)	40	70	56	52	60
15	60	44	25	70	36(a)	40	40
15	56.5(a)	42(a)	25	70	40	39(a)	40
40	70	72	60	70	48	46	56
40	88	58	60	70	47	42	51
40	70	57	60	80	60	51	71
40	86	61	55	80	55(a)	49(a)	66
40	70(a)	51(a)	43	90	51.5(a)	52	50
70	80(a)	78	80	90	60.5	51(a)	60
70	107	81	80	90	58	51	60
70	112	66(a)	80	100	68	61	75
75	93.5(a)	68(a)	61	100	58(a)	50(a)	66
75	114	79	75	120	56(a)	57(a)	65
100	118.5	77	77	120	64	57	60
100	100(a)	69(a)	68	120	74	65	60

Table B.11-1 Charpy V-Notch Test Data for the Intermediate Shell Plate, Heat # A9154-1 (cont.)

Longitudinal				Transverse			
Temp. (°F)	CVN Impact Energy (ft-lb)	Lateral Expansion (mils)	Shear (%)	Temp. (°F)	CVN Impact Energy (ft-lb)	Lateral Expansion (mils)	Shear (%)
125	127[(a)]	84[(a)]	100	125	69[(a)(b)]	61[(a)]	98
150	126.5[(a)]	86	100	150	82[(a)]	69[(a)]	100
150	131	85	100	210	76	60	100
210	140.5	84	100	210	70.5[(a)]	58[(a)(b)]	100
210	132.5[(a)]	83[(a)]	100	210	72	58	100
212	121[(a)(b)]	96	99	212	82.5	70	100
212	144	97	99	212	76.5[(a)]	69	100
212	144	92[(a)(b)]	99	212	83	68[(a)]	100

Notes for Table B.11-1:

(a) Minimum value used in the CVGRAPH plots in accordance with ASME Code III Subarticle NB-2331 criteria.

(b) The value fixed as the upper shelf in CVGRAPH plots.

Table B.11-2 Drop-Weight Test Data for Intermediate Shell Plate, Heat # A9154-1

Test Temperature (°F)	Drop-Weights[(a)]	T_{NDT} (°F)
-10	2–NF	-20
-20	1–F	
-30	1–F	
-40	1–F	

Note for Table B.11-2:

(a) NF = "No Fail," F = "Fail".

B.11.1 Determination of the Initial RT_{NDT}

Using the data summarized in Tables B.11-1 and B.11-2, the initial RT_{NDT} value can be determined in accordance with the ASME Code Section III, Subarticle NB-2331 requirements. Following the requirements of ASME Code Section III, Subarticle NB-2331, Paragraph (a)(2), the minimum Charpy V-notch test data is first checked at a temperature not greater than the drop-weight T_{NDT} (or NDT) plus 60°F to determine if the material exhibits at least 50 ft-lb absorbed energy and 35 mils LE in the "weak" direction. Charpy V-notch tests were conducted at 40°F, T_{NDT} + 60°F (-20°F + 60°F = 40°F). The minimum Charpy V-notch test data at this temperature did **NOT** exhibit a minimum of 50 ft-lb absorbed energy and 35 mils lateral expansion; therefore, the Charpy V-notch tests at T_{NDT} + 60°F would **NOT** satisfy the criteria.

To precisely determine the temperature at which 50 ft-lb and 35 mils LE were obtained on the specimens, the unirradiated Charpy V-notch data may be plotted and fit using a hyperbolic tangent curve-fitting software, CVGRAPH. Only the minimum data points at each Charpy V-notch test temperature were used as input to the curve-fitting software, in accordance with ASME Code Section III, Subarticle NB-2331, Paragraph (a)(4). When plotting, the USE is fixed to the minimum Charpy impact energy or lateral expansion used in the plot which experience ≥ 95% shear. The resulting CVGRAPH figures are contained in the following pages for Charpy V-notch absorbed energy and lateral expansion.

Using these figures, the temperature at which 50 ft-lb absorbed energy and 35 mils lateral expansion were achieved may be obtained. The absorbed energy test data is more conservative than the lateral expansion test data; therefore, it becomes the dominant data set in defining initial RT_{NDT}.

$T_{50\ ft\text{-}lb} = 80.8°F$
$T_{35\ mils} = 48.7°F$
$T_{Cv} = Max\ [T_{50\ ft\text{-}lb}, T_{35\ mil}] = Max\ [80.8°F, 48.7°F]$
$T_{Cv} = 80.8°F$

Following the requirements of ASME Code Section III, Subarticle NB-2331, Paragraph (a)(3), the initial RT_{NDT} is the higher of T_{NDT} (determined from the drop-weight tests) and T_{Cv} (determined above) minus 60°F.

$RT_{NDT} = Max\ [T_{NDT}, T_{CV} - 60°F]$
$RT_{NDT} = Max\ [-20°F, 80.8°F - 60°F] = Max\ [-20°F, 20.8°F]$

Intermediate Shell Plate, Heat # A9154-1 Initial RT_{NDT} = 21°F

B.11.2 Determination of the Initial USE

The current 10 CFR 50, Appendix G requirements specify that USE be calculated based on ASTM E185-82. Herein, USE is calculated based on an interpretation of ASTM E185-82 that is best explained by the most recent version of the ASTM E185 manual (2016 version). Using the guidelines in ASTM E185-82 and E185-16, the average of all Charpy data ≥ 95% shear is reported as the USE. In some instances, there may be data deemed 'out of family,' which are removed from the determination of the USE based on engineering judgment. However, the use of engineering judgment to remove 'out of family' data was not necessary for this material. The Transverse (weak direction) USE is displayed below; this value is the average of each of the impact energy values contained in Table B.11-1 with shear ≥ 95%.

Intermediate Shell Plate, Heat # A9154-1 Initial USE = Average (69 ,82, 76, 70.5, 72, 82.5, 76.5, 83) ft-lb
= 76 ft-lb

B.11.3 Chemistry

The Cu and Ni wt. % chemical compositions of the V.C. Summer Unit 1 reactor vessel materials were defined by a review of the available original test documentation. The material's chemical properties are defined as the average of all available data. When component specific data was not available, a generic value was defined as the mean plus one standard deviation of available data from similar materials. This method is consistent with Regulatory Guide 1.99, Revision 2, which allows the mean plus one standard deviation method to be used for conservative chemistry estimates based on generic data if component specific data is not available. The chemical compositions are summarized in Table B.11-3.

Table B.11-3 Chemistry Data for Intermediate Shell Plate, Heat # A9154-1

Copper (wt.-%)	**Nickel (wt.-%)**	**Source**
0.10	0.51	CMTR, Lukens Steel Analysis

Therefore, the chemical content will be defined as shown below going forward:

Intermediate Shell Plate, Heat # A9154-1 Cu Content = 0.10 wt-%

Intermediate Shell Plate, Heat # A9154-1 Ni Content = 0.51 wt-%

Figure B.11-1 Intermediate Shell Plate, Heat # A9154-1 Plot of Measured Transverse Direction CVN Data

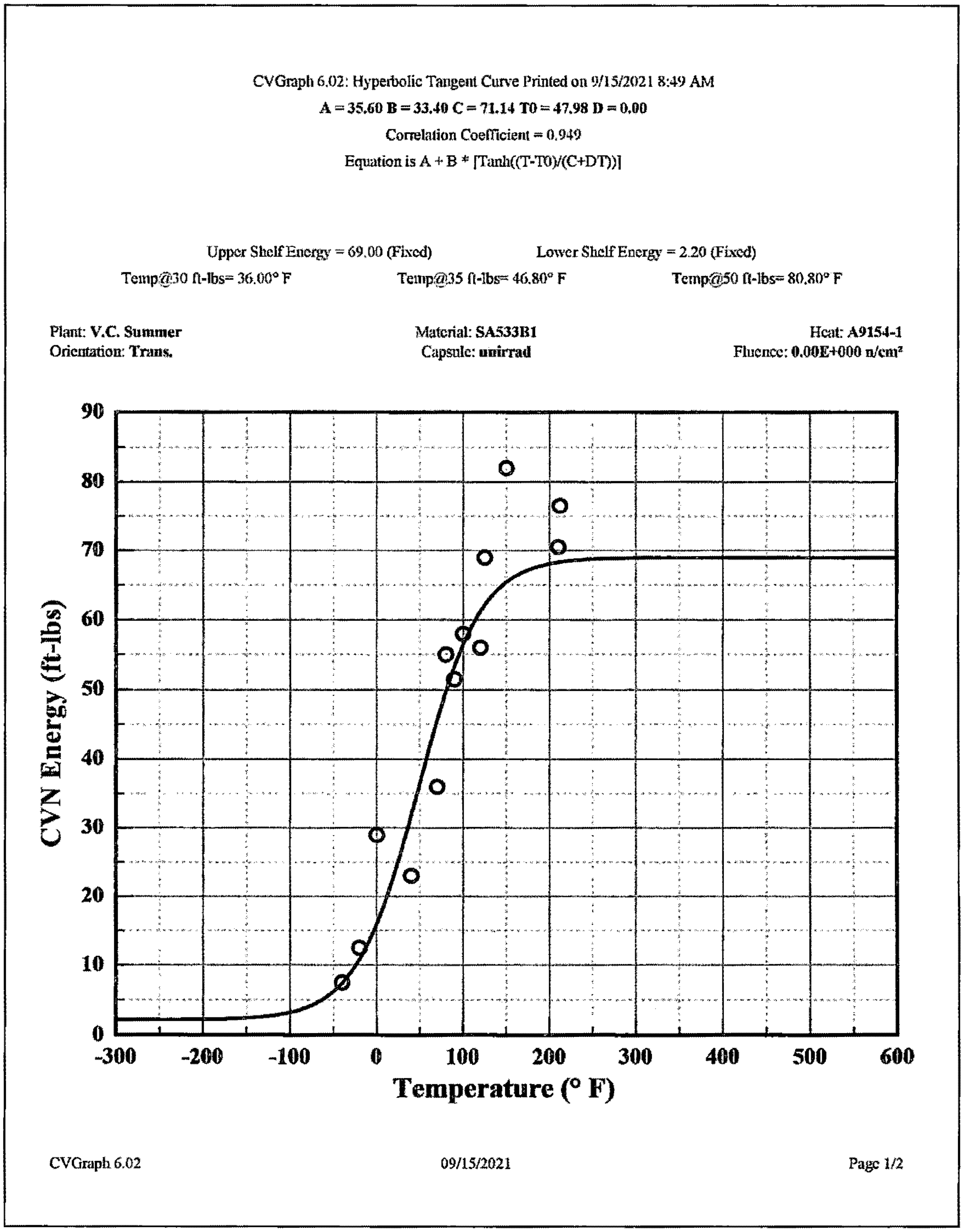


Figure B.11-1 Intermediate Shell Plate, Heat # A9154-1
Plot of Measured Transverse Direction CVN Data (cont.)

Plant: **V.C. Summer**
Orientation: **Trans.**

Material: **SA533B1**
Capsule: **unirrad**

Heat: **A9154-1**
Fluence: **0.00E+000 n/cm²**

Charpy V-Notch Data

Temperature (° F)	Input CVN	Computed CVN	Differential
-40	7.5	7.4	0.11
-20	12.5	10.8	1.69
0	29.0	16.0	13.04
40	23.0	31.9	-8.87
70	36.0	45.6	-9.62
80	55.0	49.7	5.31
90	51.5	53.3	-1.81
100	58.0	56.4	1.56
120	56.0	61.2	-5.21
125	69.0	62.1	6.87
150	82.0	65.4	16.59
210	70.5	68.3	2.20
212	76.5	68.3	8.16

CVGraph 6.02 09/15/2021 Page 2/2

Figure B.11-2 Intermediate Shell Plate, Heat # A9154-1 Plot of Measured Transverse Direction Lateral Expansion Data

CVGraph 6.02: Hyperbolic Tangent Curve Printed on 9/15/2021 8:51 AM

A = 29.50 B = 28.50 C = 62.77 T0 = 36.43 D = 0.00

Correlation Coefficient = 0.961

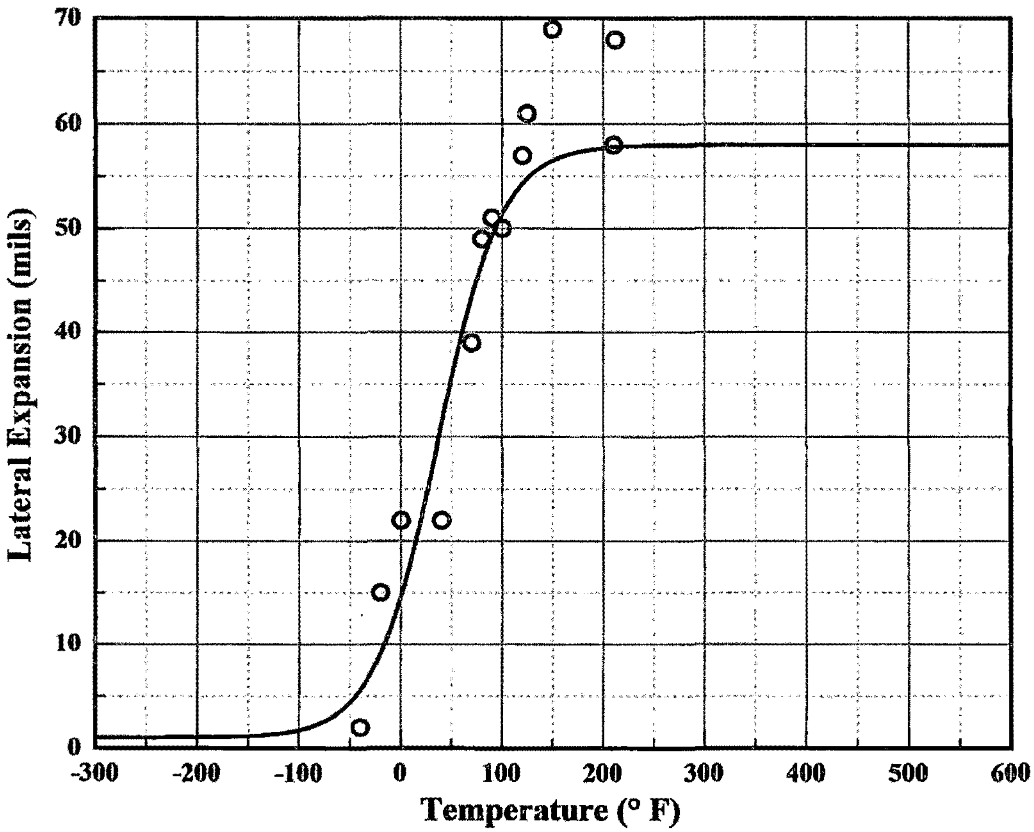
Equation is A + B * [Tanh((T-T0)/(C+DT))]

Upper Shelf L.E. = 58.00 (Fixed)   Lower Shelf L.E. = 1.00 (Fixed)

Temp@35 mils= 48.70° F

Plant: **V.C. Summer**   Material: **SA533B1**   Heat: **A9154-1**

Orientation: **Trans.**   Capsule: **unirrad**   Fluence: **0.00E+000 n/cm²**



CVGraph 6.02   09/15/2021   Page 1/2

Figure B.11-2 Intermediate Shell Plate, Heat # A9154-1
Plot of Measured Transverse Direction Lateral Expansion Data (cont.)

Plant: **V.C. Summer** Material: **SA533B1** Heat: **A9154-1**
Orientation: **Trans.** Capsule: **unirrad** Fluence: **0.00E+000 n/cm²**

Charpy V-Notch Data

Temperature (° F)	Input L. E.	Computed L. E.	Differential
-40	2.0	5.6	-3.59
-20	15.0	9.1	5.90
0	22.0	14.6	7.40
40	22.0	31.1	-9.12
70	39.0	43.4	-4.44
80	49.0	46.6	2.38
90	51.0	49.2	1.75
100	50.0	51.4	-1.36
120	57.0	54.3	2.72
125	61.0	54.8	6.20
150	69.0	56.5	12.49
210	58.0	57.8	0.23
212	68.0	57.8	10.21

CVGraph 6.02 09/15/2021 Page 2/2

B.12 V.C. Summer Unit 1 Intermediate Shell Plate, Heat # A9153-2

Tables B.12-1 and B.12-2 summarize all available Charpy V-notch test data and drop-weight test data taken from the V.C. Summer Unit 1 CMTRs for Intermediate Shell Plate, Heat # A9153-2.

Table B.12-1 Charpy V-Notch Test Data for the Intermediate Shell Plate, Heat # A9153-2

Longitudinal				Transverse			
Temp. (°F)	CVN Impact Energy (ft-lb)	Lateral Expansion (mils)	Shear (%)	Temp. (°F)	CVN Impact Energy (ft-lb)	Lateral Expansion (mils)	Shear (%)
-100	9	2[(a)]	1	-20	14[(a)]	21	20
-100	3[(a)]	6	1	-20	26	23	25
-100	8	5	1	-20	18	16[(a)]	25
-20	54[(a)]	42[(a)]	20	20	43	38	35
-20	58	45	20	20	42.5[(a)]	36	35
-20	54	44	20	20	47	35[(a)]	40
10	68	52	30	40	59	49	40
10	56[(a)]	58	30	40	51	42[(a)]	45
10	74	45[(a)]	30	40	50[(a)]	43	50
40	82	63	50	70	65[(a)]	56	70
40	83	64	50	70	71	59	80
40	82[(a)]	63[(a)]	50	70	68	55[(a)]	65
70	108	72[(a)]	60	120	93	74	80
70	107	74	60	120	90[(a)]	71[(a)]	80
70	105[(a)]	74	60	120	94	76	80
212	142	92[(a)(b)]	99	212	101[(a)(b)]	69[(a)(b)]	100
212	146	97	99	212	108	78	100
212	136[(a)(b)]	96	99	212	111.5	72	100

Notes for Table B.12-1:

(a) Minimum value used in the CVGRAPH plots in accordance with ASME Code III Subarticle NB-2331 criteria.

(b) The value fixed as the upper shelf in CVGRAPH plots.

Table B.12-2 Drop-Weight Test Data for Intermediate Shell Plate, Heat # A9153-2

Test Temperature (°F)	Drop-Weights[(a)]	T_{NDT} (°F)
10	1–NF	-20
-10	2–NF	
-20	1–F	

Note for Table B.12-2:
(a) NF = "No Fail," F = "Fail".

B.12.1 Determination of the Initial RT_{NDT}

Using the data summarized in Tables B.12-1 and B.12-2, the initial RT_{NDT} value can be determined in accordance with the ASME Code Section III, Subarticle NB-2331 requirements. Following the requirements of ASME Code Section III, Subarticle NB-2331, Paragraph (a)(2), the minimum Charpy V-notch test data is first checked at a temperature not greater than the drop-weight T_{NDT} (or NDT) plus 60°F to determine if the material exhibits at least 50 ft-lb absorbed energy and 35 mils LE in the "weak" direction. Charpy V-notch tests were conducted at 40°F, T_{NDT} + 60°F (-20°F + 60°F = 40°F). The minimum Charpy V-notch test data at this temperature exhibit a minimum of 50 ft-lb absorbed energy and 35 mils lateral expansion; therefore, the Charpy V-notch tests at T_{NDT} + 60°F satisfy the criteria. Per ASME Code Section III, Subarticle NB-2331, Paragraph (a)(2), the requirements have been met such that T_{NDT} is the initial reference temperature RT_{NDT}. However, since the Charpy V-notch tests at T_{NDT} + 60°F met the ASME Section III criterion with no margin, it was decided to verify the RT_{NDT} by plotting and fitting the unirradiated Charpy V-notch data using a hyperbolic tangent curve-fitting software, CVGRAPH. Only the minimum data points at each Charpy V-notch test temperature were used as input to the curve-fitting software, in accordance with ASME Code Section III, Subarticle NB-2331, Paragraph (a)(4). When plotting, the USE is fixed to the minimum Charpy impact energy or lateral expansion used in the plot which experience ≥ 95% shear. The resulting CVGRAPH figures are contained in the following pages for Charpy V-notch absorbed energy and lateral expansion.

Using these figures, the temperature at which 50 ft-lb absorbed energy and 35 mils lateral expansion were achieved may be obtained. The absorbed energy test data is more conservative than the lateral expansion test data; therefore, it becomes the dominant data set in defining initial RT_{NDT}.

$T_{50\ ft\text{-}lb}$ = 40.4°F
$T_{35\ mils}$ = 22.5°F
T_{Cv} = Max [$T_{50\ ft\text{-}lb}$, $T_{35\ mil}$] = Max [40.4°F, 22.5°F]
T_{Cv} = 40.4°F

Following the requirements of ASME Code Section III, Subarticle NB-2331, Paragraph (a)(3), the initial RT_{NDT} is the higher of T_{NDT} (determined from the drop-weight tests) and T_{Cv} (determined above) minus 60°F.

RT_{NDT} = Max [T_{NDT}, T_{CV} - 60°F]
RT_{NDT} = Max [-20°F, 40.4°F - 60°F] = Max [-20°F, -19.6°F]

Intermediate Shell Plate, Heat # A9153-2 Initial RT_{NDT} = -20°F

B.12.2 Determination of the Initial USE

The current 10 CFR 50, Appendix G requirements specify that USE be calculated based on ASTM E185-82. Herein, USE is calculated based on an interpretation of ASTM E185-82 that is best explained by the most recent version of the ASTM E185 manual (2016 version). Using the guidelines in ASTM E185-82 and E185-16, the average of all Charpy data ≥ 95% shear is reported as the USE. In some instances, there may be data deemed 'out of family,' which are removed from the determination of the USE based on engineering judgment. However, the use of engineering judgment to remove 'out of family' data was not necessary for this material. The Transverse (weak direction) USE is displayed below; this value is the average of each of the impact energy values contained in Table B.12-1 with shear ≥ 95%.

Intermediate Shell Plate, Heat # A9153-2 Initial USE = Average (101, 108, 111.5) ft-lb
= 107 ft-lb

B.12.3 Chemistry

The Cu and Ni wt. % chemical compositions of the V.C. Summer Unit 1 reactor vessel materials were defined by a review of the available original test documentation. The material's chemical properties are defined as the average of all available data. When component specific data was not available, a generic value was defined as the mean plus one standard deviation of available data from similar materials. This method is consistent with Regulatory Guide 1.99, Revision 2, which allows the mean plus one standard deviation method to be used for conservative chemistry estimates based on generic data if component specific data is not available. The chemical compositions are summarized in Table B.12-3.

Table B.12-3 Chemistry Data for Intermediate Shell Plate, Heat # A9153-2

Copper (wt.-%)	Nickel (wt.-%)	Source
0.09	0.45	CMTR, Lukens Steel Analysis

Therefore, the chemical content will be defined as shown below going forward:

Intermediate Shell Plate, Heat # A9153-2 Cu Content = 0.09 wt-%

Intermediate Shell Plate, Heat # A9153-2 Ni Content = 0.45 wt-%

**Figure B.12-1 Intermediate Shell Plate, Heat # A9153-2
Plot of Measured Transverse Direction CVN Data**

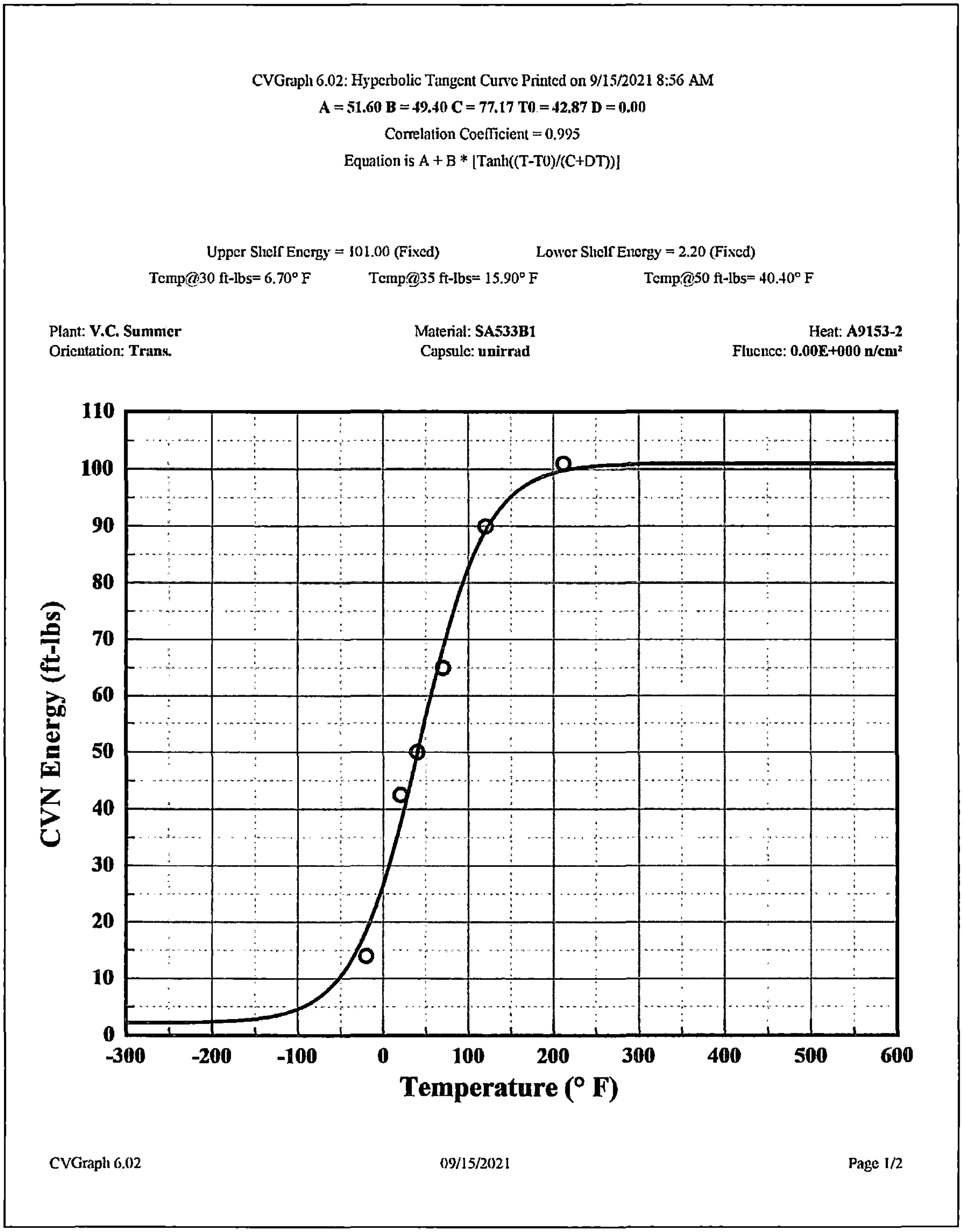


Figure B.12-1 Intermediate Shell Plate, Heat # A9153-2 Plot of Measured Transverse Direction CVN Data (cont.)

Plant: **V.C. Summer**
Orientation: **Trans.**

Material: **SA533B1**
Capsule: **unirrad**

Heat: **A9153-2**
Fluence: **0.00E+000 n/cm²**

Charpy V-Notch Data

Temperature (° F)	Input CVN	Computed CVN	Differential
-20	14.0	18.4	-4.40
20	42.5	37.4	5.12
40	50.0	49.8	0.23
70	65.0	68.3	-3.29
120	90.0	89.2	0.79
212	101.0	99.8	1.22

CVGraph 6.02 09/15/2021 Page 2/2

Figure B.12-2 Intermediate Shell Plate, Heat # A9153-2 Plot of Measured Transverse Direction Lateral Expansion Data

CVGraph 6.02: Hyperbolic Tangent Curve Printed on 9/15/2021 11:28 AM

A = 51.10 B = 48.90 C = 90.58 T0 = 67.18 D = 0.00

Correlation Coefficient = 0.996

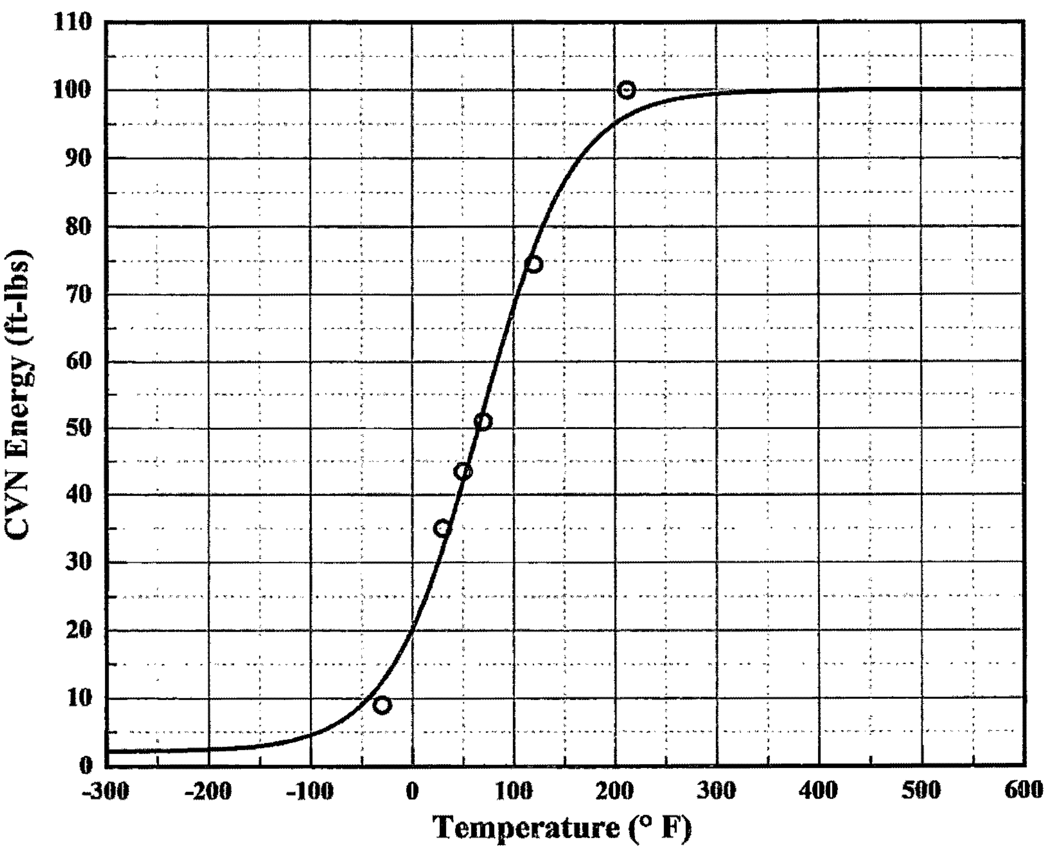
Equation is A + B * [Tanh((T-T0)/(C+DT))]

Upper Shelf Energy = 100.00 (Fixed) Lower Shelf Energy = 2.20 (Fixed)

Temp@30 ft-lbs= 25.40° F Temp@35 ft-lbs= 36.30° F Temp@50 ft-lbs= 65.20° F

Plant: **V.C. Summer** Material: **SA533B1** Heat: **C9923-1**

Orientation: **Trans.** Capsule: **unirrad** Fluence: **0.00E+000 n/cm²**



CVGraph 6.02 09/15/2021 Page 1/2

Figure B.12-2 Intermediate Shell Plate, Heat # A9153-2 Plot of Measured Transverse Direction Lateral Expansion Data (cont.)

Plant: **V.C. Summer**
Orientation: **Trans.**

Material: **SA533B1**
Capsule: **unirrad**

Heat: **A9153-2**
Fluence: **0.00E+000 n/cm²**

Charpy V-Notch Data

Temperature (° F)	Input L. E.	Computed L. E.	Differential
-20	16.0	15.4	0.61
20	35.0	33.7	1.31
40	42.0	44.0	-1.99
70	55.0	56.3	-1.29
120	71.0	65.8	5.17
212	69.0	68.8	0.19

CVGraph 6.02 09/15/2021 Page 2/2

B.13 V.C. Summer Unit 1 Lower Shell Plate, Heat # C9923-1

Tables B.13-1 and B.13-2 summarize all available Charpy V-notch test data and drop-weight test data taken from the V.C. Summer Unit 1 CMTRs for Lower Shell Plate, Heat # C9923-1.

Table B.13-1 Charpy V-Notch Test Data for the Lower Shell Plate, Heat # C9923-1

Longitudinal				Transverse			
Temp. (°F)	CVN Impact Energy (ft-lb)	Lateral Expansion (mils)	Shear (%)	Temp. (°F)	CVN Impact Energy (ft-lb)	Lateral Expansion (mils)	Shear (%)
-100	7(a)	5(a)	10	-30	11	9	10
-100	10	5	10	-30	9(a)	8(a)	5
-100	9	6	10	-30	12	11	10
-75	35	25	20	30	37	34	30
-75	17	9(a)	20	30	42.5	37	35
-75	12(a)	15	20	30	35(a)	34(a)	30
-50	52	40	50	50	47	43	40
-50	60	45	50	50	44	40	30
-50	44(a)	38(a)	50	50	43.5(a)	39(a)	35
10	92	70	80	70	68	57	55
10	70(a)	64	80	70	51	47	50
10	82	61(a)	80	70	51(a)	47(a)	50
40	92	77	80	120	74.5(a)	63(a)	60
40	100	76(a)	80	120	81	64	70
40	89(a)	78	80	120	80	63	70
212	148	96	99	212	104	83	100
212	147(a)(b)	96(a)(b)	99	212	114	81	95
212	150	96	99	212	100(a)(b)	77(a)(b)	95

Notes for Table B.13-1:

(a) Minimum value used in the CVGRAPH plots in accordance with ASME Code III Subarticle NB-2331 criteria.

(b) The value fixed as the upper shelf in CVGRAPH plots.

Table B.13-2 Drop-Weight Test Data for Lower Shell Plate, Heat # C9923-1

Test Temperature (°F)	Drop-Weights[(a)]	T_{NDT} (°F)
10	1–NF	-30
0	1–NF	
-20	2–NF	
-30	1–F	
-40	1–F	

Note for Table B.13-2:
(a) NF = "No Fail," F = "Fail".

B.13.1 Determination of the Initial RT_{NDT}

Using the data summarized in Tables B.13-1 and B.13-2, the initial RT_{NDT} value can be determined in accordance with the ASME Code Section III, Subarticle NB-2331 requirements. Following the requirements of ASME Code Section III, Subarticle NB-2331, Paragraph (a)(2), the minimum Charpy V-notch test data is first checked at a temperature not greater than the drop-weight T_{NDT} (or NDT) plus 60°F to determine if the material exhibits at least 50 ft-lb absorbed energy and 35 mils LE in the "weak" direction. Charpy V-notch tests were conducted at 30°F, T_{NDT} + 60°F (-30°F + 60°F = 30°F). The minimum Charpy V-notch test data at this temperature did **NOT** exhibit a minimum of 50 ft-lb absorbed energy and 35 mils lateral expansion; therefore, the Charpy V-notch tests at T_{NDT} + 60°F would **NOT** satisfy the criteria.

To precisely determine the temperature at which 50 ft-lb and 35 mils LE were obtained on the specimens, the unirradiated Charpy V-notch data may be plotted and fit using a hyperbolic tangent curve-fitting software, CVGRAPH. Only the minimum data points at each Charpy V-notch test temperature were used as input to the curve-fitting software, in accordance with ASME Code Section III, Subarticle NB-2331, Paragraph (a)(4). When plotting, the USE is fixed to the minimum Charpy impact energy or lateral expansion used in the plot which experience ≥ 95% shear. The resulting CVGRAPH figures are contained in the following pages for Charpy V-notch absorbed energy and lateral expansion.

Using these figures, the temperature at which 50 ft-lb absorbed energy and 35 mils lateral expansion were achieved may be obtained. The absorbed energy test data is more conservative than the lateral expansion test data; therefore, it becomes the dominant data set in defining initial RT_{NDT}.

$T_{50\ ft\text{-}lb} = 65.2°F$
$T_{35\ mils} = 40.5°F$
$T_{Cv} = Max\ [T_{50\ ft\text{-}lb}, T_{35\ mil}] = Max\ [65.2°F, 40.5°F]$
$T_{Cv} = 65.2°F$

Following the requirements of ASME Code Section III, Subarticle NB-2331, Paragraph (a)(3), the initial RT_{NDT} is the higher of T_{NDT} (determined from the drop-weight tests) and T_{Cv} (determined above) minus 60°F.

RT_{NDT} = Max [T_{NDT}, T_{CV} - 60°F]
RT_{NDT} = Max [-30°F, 65.2°F - 60°F] = Max [-30°F, 5.2°F]

Lower Shell Plate, Heat # C9923-1 Initial RT_{NDT} = 5°F

B.13.2 Determination of the Initial USE

The current 10 CFR 50, Appendix G requirements specify that USE be calculated based on ASTM E185-82. Herein, USE is calculated based on an interpretation of ASTM E185-82 that is best explained by the most recent version of the ASTM E185 manual (2016 version). Using the guidelines in ASTM E185-82 and E185-16, the average of all Charpy data $\geq$ 95% shear is reported as the USE. In some instances, there may be data deemed 'out of family,' which are removed from the determination of the USE based on engineering judgment. However, the use of engineering judgment to remove 'out of family' data was not necessary for this material. The Transverse (weak direction) USE is displayed below; this value is the average of each of the impact energy values contained in Table B.13-1 with shear $\geq$ 95%.

Lower Shell Plate, Heat # C9923-1 Initial USE = Average (104, 114, 100) ft-lb
= 106 ft-lb

B.13.3 Chemistry

The Cu and Ni wt. % chemical compositions of the V.C. Summer Unit 1 reactor vessel materials were defined by a review of the available original test documentation. The material's chemical properties are defined as the average of all available data. When component specific data was not available, a generic value was defined as the mean plus one standard deviation of available data from similar materials. This method is consistent with Regulatory Guide 1.99, Revision 2, which allows the mean plus one standard deviation method to be used for conservative chemistry estimates based on generic data if component specific data is not available. The chemical compositions are summarized in Table B.13-3.

Table B.13-3 Chemistry Data for Lower Shell Plate, Heat # C9923-1

Copper (wt.-%)	Nickel (wt.-%)	Source
0.08	0.41	CMTR, Lukens Steel Analysis

Therefore, the chemical content will be defined as shown below going forward:

Lower Shell Plate, Heat # C9923-1 Cu Content = 0.08 wt-%

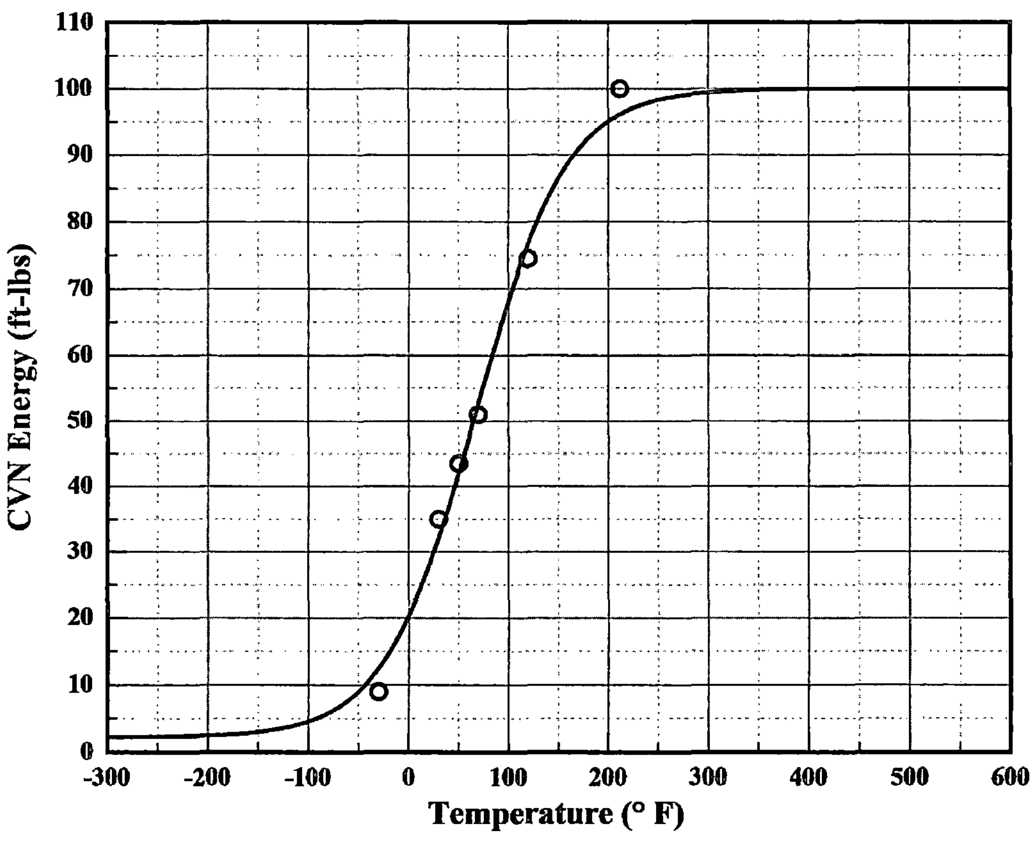
Lower Shell Plate, Heat # C9923-1 Ni Content = 0.41 wt-%

Figure B.13-1 Lower Shell Plate, Heat # C9923-1
Plot of Measured Transverse Direction CVN Data

CVGraph 6.02: Hyperbolic Tangent Curve Printed on 9/15/2021 11:28 AM
A = 51.10 B = 48.90 C = 90.58 T0 = 67.18 D = 0.00
Correlation Coefficient = 0.996
Equation is A + B * [Tanh((T-T0)/(C+DT))]

Upper Shelf Energy = 100.00 (Fixed) Lower Shelf Energy = 2.20 (Fixed)
Temp@30 ft-lbs= 25.40° F Temp@35 ft-lbs= 36.30° F Temp@50 ft-lbs= 65.20° F

Plant: **V.C. Summer** Material: **SA533B1** Heat: **C9923-1**
Orientation: **Trans.** Capsule: **unirrad** Fluence: **0.00E+000 n/cm²**



CVGraph 6.02 09/15/2021 Page 1/2

Figure B.13-1 Lower Shell Plate, Heat # C9923-1
Plot of Measured Transverse Direction CVN Data (cont.)

Plant: **V.C. Summer**
Orientation: **Trans.**

Material: **SA533B1**
Capsule: **unirrad**

Heat: **C9923-1**
Fluence: **0.00E+000 n/cm²**

Charpy V-Notch Data

Temperature (° F)	Input CVN	Computed CVN	Differential
-30	9.0	12.4	-3.44
30	35.0	32.1	2.91
50	43.5	41.9	1.56
70	51.0	52.6	-1.62
120	74.5	76.8	-2.27
212	100.0	96.2	3.84

CVGraph 6.02 09/15/2021 Page 2/2

Figure B.13-2 Lower Shell Plate, Heat # C9923-1
Plot of Measured Transverse Direction Lateral Expansion Data

CVGraph 6.02: Hyperbolic Tangent Curve Printed on 9/15/2021 11:30 AM

A = 39.00 B = 38.00 C = 85.82 T0 = 49.56 D = 0.00

Correlation Coefficient = 0.995

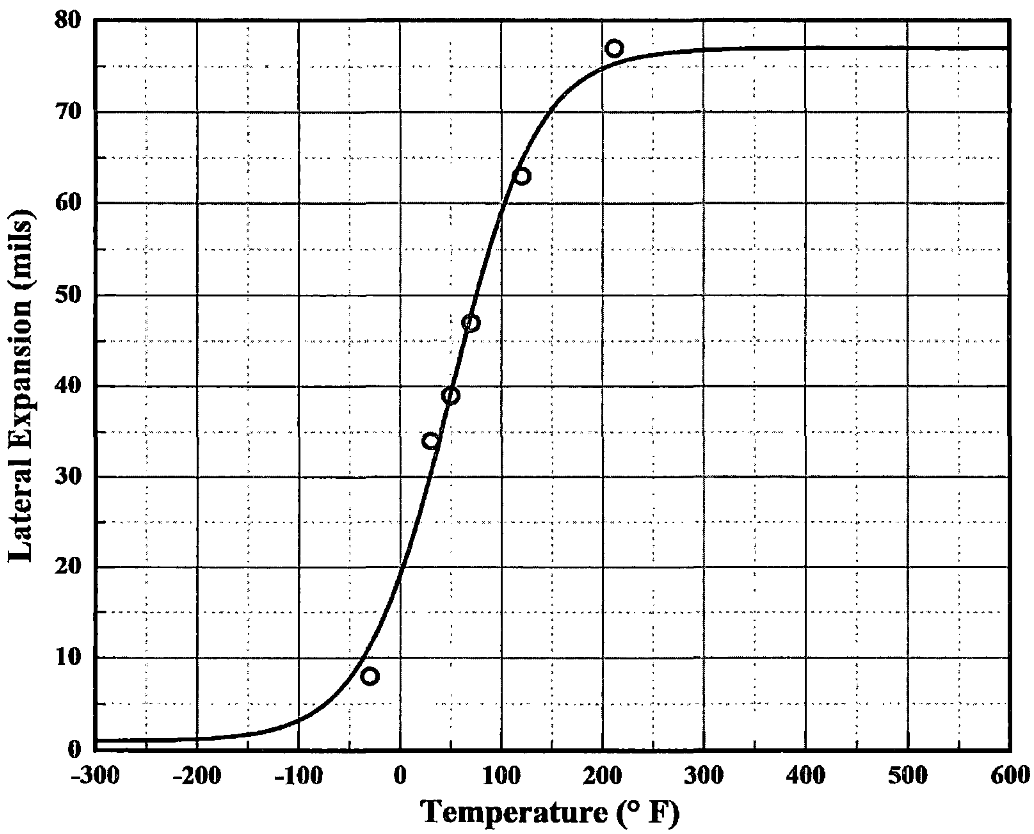
Equation is A + B * [Tanh((T-T0)/(C+DT))]

Upper Shelf L.E. = 77.00 (Fixed) Lower Shelf L.E. = 1.00 (Fixed)

Temp@35 mils= 40.50° F

Plant: **V.C. Summer** Material: **SA533B1** Heat: **C9923-1**

Orientation: **Trans.** Capsule: **unirrad** Fluence: **0.00E+000 n/cm²**



CVGraph 6.02 09/15/2021 Page 1/2

Figure B.13-2 Lower Shell Plate, Heat # C9923-1
Plot of Measured Transverse Direction Lateral Expansion Data (cont.)

Plant: **V.C. Summer**
Orientation: **Trans.**

Material: **SA533B1**
Capsule: **unirrad**

Heat: **C9923-1**
Fluence: **0.00E+000 n/cm²**

Charpy V-Notch Data

Temperature (° F)	Input L. E.	Computed L. E.	Differential
-30	8.0	11.3	-3.29
30	34.0	30.5	3.51
50	39.0	39.2	-0.20
70	47.0	47.9	-0.89
120	63.0	64.7	-1.67
212	77.0	75.3	1.69

CVGraph 6.02 09/15/2021 Page 2/2

B.14 V.C. Summer Unit 1 Lower Shell Plate, Heat # C9923-2

Tables B.14-1 and B.14-2 summarize all available Charpy V-notch test data and drop-weight test data taken from the V.C. Summer Unit 1 CMTRs for Lower Shell Plate, Heat # C9923-2.

Table B.14-1 Charpy V-Notch Test Data for the Lower Shell Plate, Heat # C9923-2

Longitudinal				Transverse			
Temp. (°F)	CVN Impact Energy (ft-lb)	Lateral Expansion (mils)	Shear (%)	Temp. (°F)	CVN Impact Energy (ft-lb)	Lateral Expansion (mils)	Shear (%)
-100	4(a)	3	1	-10	23(a)	27	25
-100	5	2(a)	1	-10	31	28	20
-100	5	2	1	-10	33.5	29	25
-50	8	10	1	40	56	48	50
-50	6(a)	7(a)	1	40	42.5	37	35
-50	12	9	1	40	37.5(a)	34(a)	30
-20	42	30(a)	40	50	40(a)	38(a)	30
-20	50	37	40	50	46	42	45
-20	35(a)	39	40	50	53	45	45
10	56	44(a)	50	70	55	50	55
10	56	46	50	70	50(a)	45(a)	50
10	55(a)	46	50	70	51	46	55
40	86	64	70	120	82	68	80
40	76(a)	61(a)	70	120	79(a)	65(a)	75
40	85	66	70	120	84	74	85
212	164	95	99	212	93	76(a)(b)	100
212	164	95(a)(b)	99	212	94	78	100
212	155(a)(b)	96	99	212	88(a)(b)	77	100

Notes for Table B.14-1:

(a) Minimum value used in the CVGRAPH plots in accordance with ASME Code III Subarticle NB-2331 criteria.

(b) The value fixed as the upper shelf in CVGRAPH plots.

Table B.14-2 Drop-Weight Test Data for Lower Shell Plate, Heat # C9923-2

Test Temperature (°F)	Drop-Weights[(a)]	T_{NDT} (°F)
10	1–NF	-10
0	2–NF	
-10	1–F	
-20	1–F	
-20	1–F	

Note for Table B.14-2:
(a) NF = "No Fail," F = "Fail".

B.14.1 Determination of the Initial RT_{NDT}

Using the data summarized in Tables B.14-1 and B.14-2, the initial RT_{NDT} value can be determined in accordance with the ASME Code Section III, Subarticle NB-2331 requirements. Following the requirements of ASME Code Section III, Subarticle NB-2331, Paragraph (a)(2), the minimum Charpy V-notch test data is first checked at a temperature not greater than the drop-weight T_{NDT} (or NDT) plus 60°F to determine if the material exhibits at least 50 ft-lb absorbed energy and 35 mils LE in the "weak" direction. Charpy V-notch tests were conducted at 50°F, T_{NDT} + 60°F (-10°F + 60°F = 50°F). The minimum Charpy V-notch test data at this temperature did **NOT** exhibit a minimum of 50 ft-lb absorbed energy and 35 mils lateral expansion; therefore, the Charpy V-notch tests at T_{NDT} + 60°F would **NOT** satisfy the criteria.

To precisely determine the temperature at which 50 ft-lb and 35 mils LE were obtained on the specimens, the unirradiated Charpy V-notch data may be plotted and fit using a hyperbolic tangent curve-fitting software, CVGRAPH. Only the minimum data points at each Charpy V-notch test temperature were used as input to the curve-fitting software, in accordance with ASME Code Section III, Subarticle NB-2331, Paragraph (a)(4). When plotting, the USE is fixed to the minimum Charpy impact energy or lateral expansion used in the plot which experience ≥ 95% shear. The resulting CVGRAPH figures are contained in the following pages for Charpy V-notch absorbed energy and lateral expansion.

Using these figures, the temperature at which 50 ft-lb absorbed energy and 35 mils lateral expansion were achieved may be obtained. The absorbed energy test data is more conservative than the lateral expansion test data; therefore, it becomes the dominant data set in defining initial RT_{NDT}.

$T_{50\ ft\text{-}lb}$ = 63.6°F
$T_{35\ mils}$ = 33.9°F
T_{Cv} = Max [$T_{50\ ft\text{-}lb}$, $T_{35\ mil}$] = Max [63.6°F, 33.9°F]
T_{Cv} = 63.6°F

Following the requirements of ASME Code Section III, Subarticle NB-2331, Paragraph (a)(3), the initial RT_{NDT} is the higher of T_{NDT} (determined from the drop-weight tests) and T_{Cv} (determined above) minus 60°F.

RT_{NDT} = Max [T_{NDT}, T_{CV} - 60°F]
RT_{NDT} = Max [-10°F, 63.6°F - 60°F] = Max [-10°F, 3.6°F]

Lower Shell Plate, Heat # C9923-2 Initial RT_{NDT} = 4°F

B.14.2 Determination of the Initial USE

The current 10 CFR 50, Appendix G requirements specify that USE be calculated based on ASTM E185-82. Herein, USE is calculated based on an interpretation of ASTM E185-82 that is best explained by the most recent version of the ASTM E185 manual (2016 version). Using the guidelines in ASTM E185-82 and E185-16, the average of all Charpy data ≥ 95% shear is reported as the USE. In some instances, there may be data deemed 'out of family,' which are removed from the determination of the USE based on engineering judgment. However, the use of engineering judgment to remove 'out of family' data was not necessary for this material. The Transverse (weak direction) USE is displayed below; this value is the average of each of the impact energy values contained in Table B.14-1 with shear ≥ 95%.

Lower Shell Plate, Heat # C9923-2 Initial USE = Average (93, 94, 88) ft-lb
= 92 ft-lb

B.14.3 Chemistry

The Cu and Ni wt. % chemical compositions of the V.C. Summer Unit 1 reactor vessel materials were defined by a review of the available original test documentation. The material's chemical properties are defined as the average of all available data. When component specific data was not available, a generic value was defined as the mean plus one standard deviation of available data from similar materials. This method is consistent with Regulatory Guide 1.99, Revision 2, which allows the mean plus one standard deviation method to be used for conservative chemistry estimates based on generic data if component specific data is not available. The chemical compositions are summarized in Table B.14-3.

Table B.14-3 Chemistry Data for Lower Shell Plate, Heat # C9923-2

Copper (wt.-%)	Nickel (wt.-%)	Source
0.08	0.41	CMTR, Lukens Steel Analysis

Therefore, the chemical content will be defined as shown below going forward:

Lower Shell Plate, Heat # C9923-2 Cu Content = 0.08 wt-%

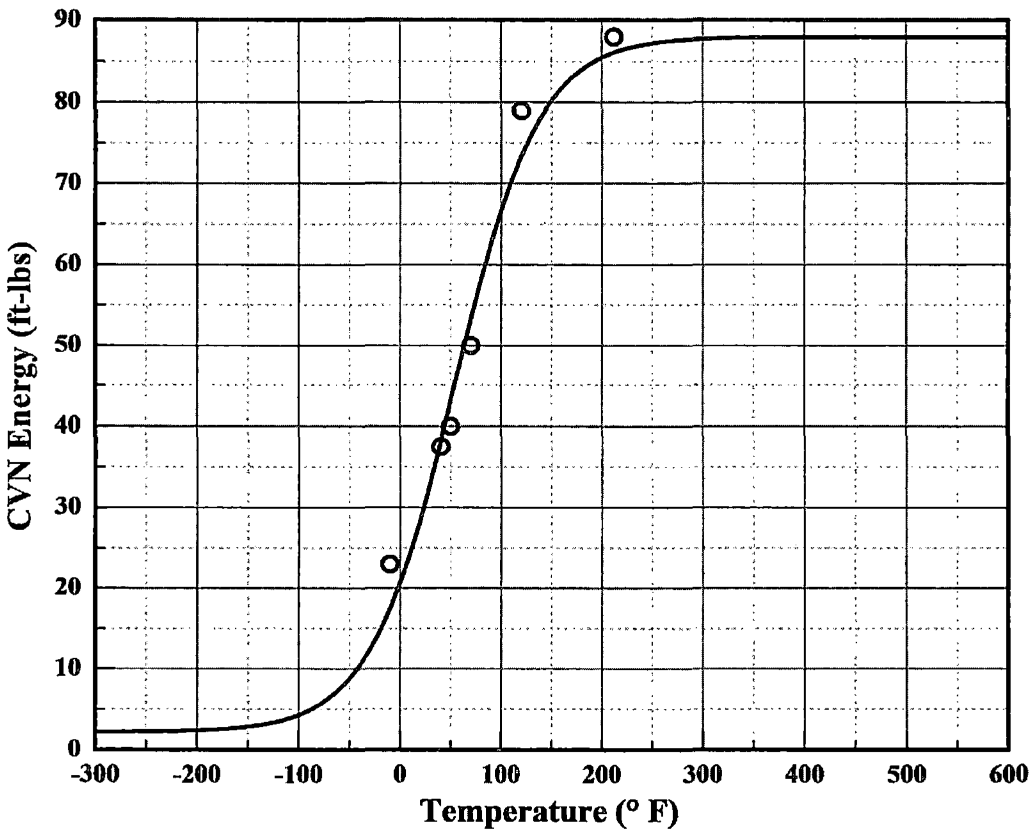
Lower Shell Plate, Heat # C9923-2 Ni Content = 0.41 wt-%

Figure B.14-1 Lower Shell Plate, Heat # C9923-2
Plot of Measured Transverse Direction CVN Data

CVGraph 6.02: Hyperbolic Tangent Curve Printed on 9/15/2021 11:33 AM
A = 45.10 B = 42.90 C = 83.59 T0 = 53.99 D = 0.00
Correlation Coefficient = 0.988
Equation is A + B * [Tanh((T-T0)/(C+DT))]

Upper Shelf Energy = 88.00 (Fixed) Lower Shelf Energy = 2.20 (Fixed)
Temp@30 ft-lbs= 23.30° F Temp@35 ft-lbs= 34.00° F Temp@50 ft-lbs= 63.60° F

Plant: **V.C. Summer** Material: **SA533B1** Heat: **C9923-2**
Orientation: **Trans.** Capsule: **unirrad** Fluence: **0.00E+000 n/cm²**



CVGraph 6.02 09/15/2021 Page 1/2

Figure B.14-1 Lower Shell Plate, Heat # C9923-2
Plot of Measured Transverse Direction CVN Data (cont.)

Plant: **V.C. Summer**
Orientation: **Trans.**

Material: **SA533B1**
Capsule: **unirrad**

Heat: **C9923-2**
Fluence: **0.00E+000 n/cm²**

Charpy V-Notch Data

Temperature (° F)	Input CVN	Computed CVN	Differential
-10	23.0	17.5	5.54
50	40.0	43.1	-3.05
40	37.5	38.0	-0.49
70	50.0	53.2	-3.22
120	79.0	73.3	5.66
212	88.0	86.1	1.91

CVGraph 6.02 09/15/2021 Page 2/2

Figure B.14-2 Lower Shell Plate, Heat # C9923-2
Plot of Measured Transverse Direction Lateral Expansion Data

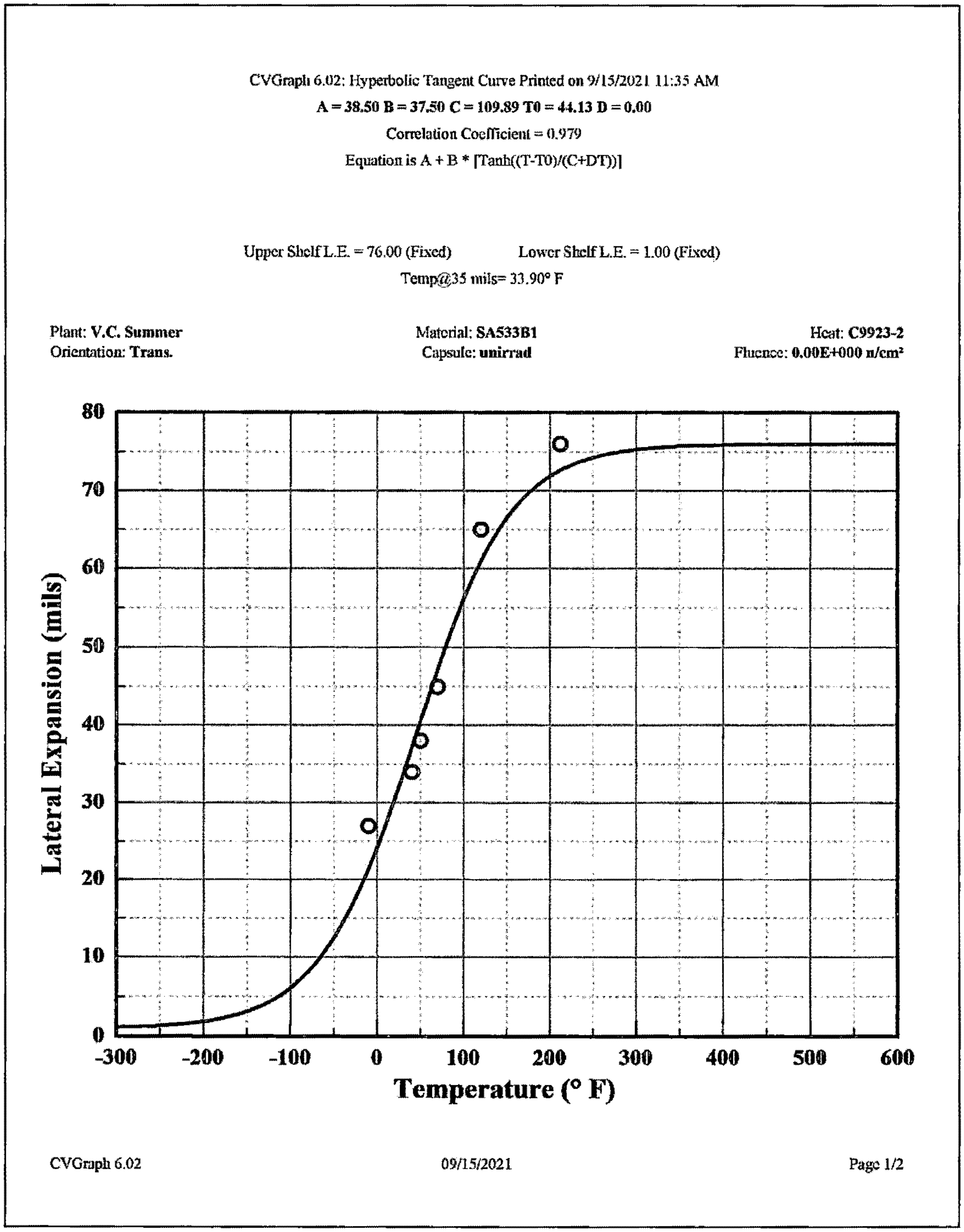


Figure B.14-2 Lower Shell Plate, Heat # C9923-2
Plot of Measured Transverse Direction Lateral Expansion Data (cont.)

Plant: **V.C. Summer**
Orientation: **Trans.**

Material: **SA533B1**
Capsule: **unirrad**

Heat: **C9923-2**
Fluence: **0.00E+000 n/cm²**

Charpy V-Notch Data

Temperature (° F)	Input L. E.	Computed L. E.	Differential
-10	27.0	21.4	5.61
50	38.0	40.5	-2.50
40	34.0	37.1	-3.09
70	45.0	47.2	-2.17
120	65.0	60.9	4.07
212	76.0	72.6	3.37

CVGraph 6.02 09/15/2021 Page 2/2

B.15 V.C. Summer Unit 1 Transition Ring Plates, Heat # A9249-1

Tables B.15-1 and B.15-2 summarize all available Charpy V-notch test data and drop-weight test data taken from the V.C. Summer Unit 1 CMTRs for the Transition Ring Plates.

Table B.15-1 Charpy V-Notch Test Data for the Transition Ring Plates, Heat # A9249-1

Longitudinal				Transverse			
Temp. (°F)	CVN Impact Energy (ft-lb)	Lateral Expansion (mils)	Shear (%)	Temp. (°F)	CVN Impact Energy (ft-lb)	Lateral Expansion (mils)	Shear (%)
-100	6	6	10	-100	11	6	10
-100	6	6	10	-100	6(a)	3(a)	10
-100	6(a)	6(a)	10	-100	7	4	10
-50	38	29	20	-50	28(a)	27(a)	30
-50	21	23(a)	20	-50	30	29	30
-50	14(a)	32	20	-50	32	31	30
-20	63	46	40	-20	54	43	40
-20	46(a)	39(a)	40	-20	36(a)	36(a)	40
-20	60	51	40	-20	39	46	40
10	96	65(a)	70	10	60	51	50
10	82	80	70	10	50	45	50
10	80(a)	67	70	10	45(a)	45(a)	50
40	115	73	80	40	61	46(a)	50
40	88(a)	72(a)	80	40	58(a)	50	50
40	108	84	80	40	61	52	50
212	163	93	99	212	108	88	99
212	160	91(a)(b)	99	212	103(a)(b)	80(a)(b)	99
212	126(a)(b)	92	99	212	111	81	99

Notes for Table B.15-1:

(a) Minimum value used in the CVGRAPH plots in accordance with ASME Code III Subarticle NB-2331 criteria.

(b) The value fixed as the upper shelf in CVGRAPH plots.

Table B.15-2 Drop-Weight Test Data for Transition Ring Plates, Heat # A9249-1

Test Temperature (°F)	Drop-Weights[(a)]	T_{NDT} (°F)
-20	1–NF	-40
-30	2–NF	
-40	1–F	

Note for Table B.15-2:
(a) NF = "No Fail," F = "Fail".

B.15.1 Determination of the Initial RT_{NDT}

Using the data summarized in Tables B.15-1 and B.15-2, the initial RT_{NDT} value can be determined in accordance with the ASME Code Section III, Subarticle NB-2331 requirements. Following the requirements of ASME Code Section III, Subarticle NB-2331, Paragraph (a)(2), the minimum Charpy V-notch test data is first checked at a temperature not greater than the drop-weight T_{NDT} (or NDT) plus 60°F to determine if the material exhibits at least 50 ft-lb absorbed energy and 35 mils LE in the "weak" direction. Charpy V-notch tests were conducted at 10°F, which is less than T_{NDT} + 60°F (-40°F + 60°F = 20°F). The minimum Charpy V-notch test data at this temperature did **NOT** exhibit a minimum of 50 ft-lb absorbed energy and 35 mils lateral expansion; therefore, the Charpy V-notch tests at T_{NDT} + 60°F would **NOT** satisfy the criteria.

To precisely determine the temperature at which 50 ft-lb and 35 mils LE were obtained on the specimens, the unirradiated Charpy V-notch data may be plotted and fit using a hyperbolic tangent curve-fitting software, CVGRAPH. Only the minimum data points at each Charpy V-notch test temperature were used as input to the curve-fitting software, in accordance with ASME Code Section III, Subarticle NB-2331, Paragraph (a)(4). When plotting, the USE is fixed to the minimum Charpy impact energy or lateral expansion used in the plot which experience ≥ 95% shear. The resulting CVGRAPH figures are contained in the following pages for Charpy V-notch absorbed energy and lateral expansion.

Using these figures, the temperature at which 50 ft-lb absorbed energy and 35 mils lateral expansion were achieved may be obtained. The absorbed energy test data is more conservative than the lateral expansion test data; therefore, it becomes the dominant data set in defining initial RT_{NDT}.

$T_{50\ ft\text{-}lb}$ = 17.9°F
$T_{35\ mils}$ = -11.2°F
T_{Cv} = Max [$T_{50\ ft\text{-}lb}$, $T_{35\ mil}$] = Max [17.9°F, -11.2°F]
T_{Cv} = 17.9°F

Following the requirements of ASME Code Section III, Subarticle NB-2331, Paragraph (a)(3), the initial RT_{NDT} is the higher of T_{NDT} (determined from the drop-weight tests) and T_{Cv} (determined above) minus 60°F.

RT_{NDT} = Max [T_{NDT}, T_{CV} - 60°F]
RT_{NDT} = Max [-40°F, 17.9°F - 60°F] = Max [-40°F, -42.1°F]

Transition Ring Plates Initial RT_{NDT} = -40°F

B.15.2 Determination of the Initial USE

The current 10 CFR 50, Appendix G requirements specify that USE be calculated based on ASTM E185-82. Herein, USE is calculated based on an interpretation of ASTM E185-82 that is best explained by the most recent version of the ASTM E185 manual (2016 version). Using the guidelines in ASTM E185-82 and E185-16, the average of all Charpy data ≥ 95% shear is reported as the USE. In some instances, there may be data deemed 'out of family,' which are removed from the determination of the USE based on engineering judgment. However, the use of engineering judgment to remove 'out of family' data was not necessary for this material. The Transverse (weak direction) USE is displayed below; this value is the average of each of the impact energy values contained in Table B.15-1 with shear ≥ 95%.

Transition Ring Plates Initial USE = Average (108, 103, 111) ft-lb
= 107 ft-lb

B.15.3 Chemistry

The Cu and Ni wt. % chemical compositions of the V.C. Summer Unit 1 reactor vessel materials were defined by a review of the available original test documentation. The material's chemical properties are defined as the average of all available data. When component specific data was not available, a generic value was defined as the mean plus one standard deviation of available data from similar materials. This method is consistent with Regulatory Guide 1.99, Revision 2, which allows the mean plus one standard deviation method to be used for conservative chemistry estimates based on generic data if component specific data is not available. The chemical compositions are summarized in Table B.15-3.

Table B.15-3 Chemistry Data for Transition Ring Plates, Heat # A9249-1

Copper (wt.-%)	Nickel (wt.-%)	Source
-	0.53	CMTR, Lukens Steel Analysis
0.172	-	Generic value based on a mean plus one standard deviation analysis of the high copper A533, Grade B, Class 1, plate materials contained in Table G.2 of ORNL/TM-2006/530.

Therefore, the chemical content will be defined as shown below going forward:

Transition Ring Plates Cu Content = 0.172 wt-%

Transition Ring Plates Ni Content = 0.53 wt-%

Figure B.15-1 Transition Ring Plates
Plot of Measured Transverse Direction CVN Data

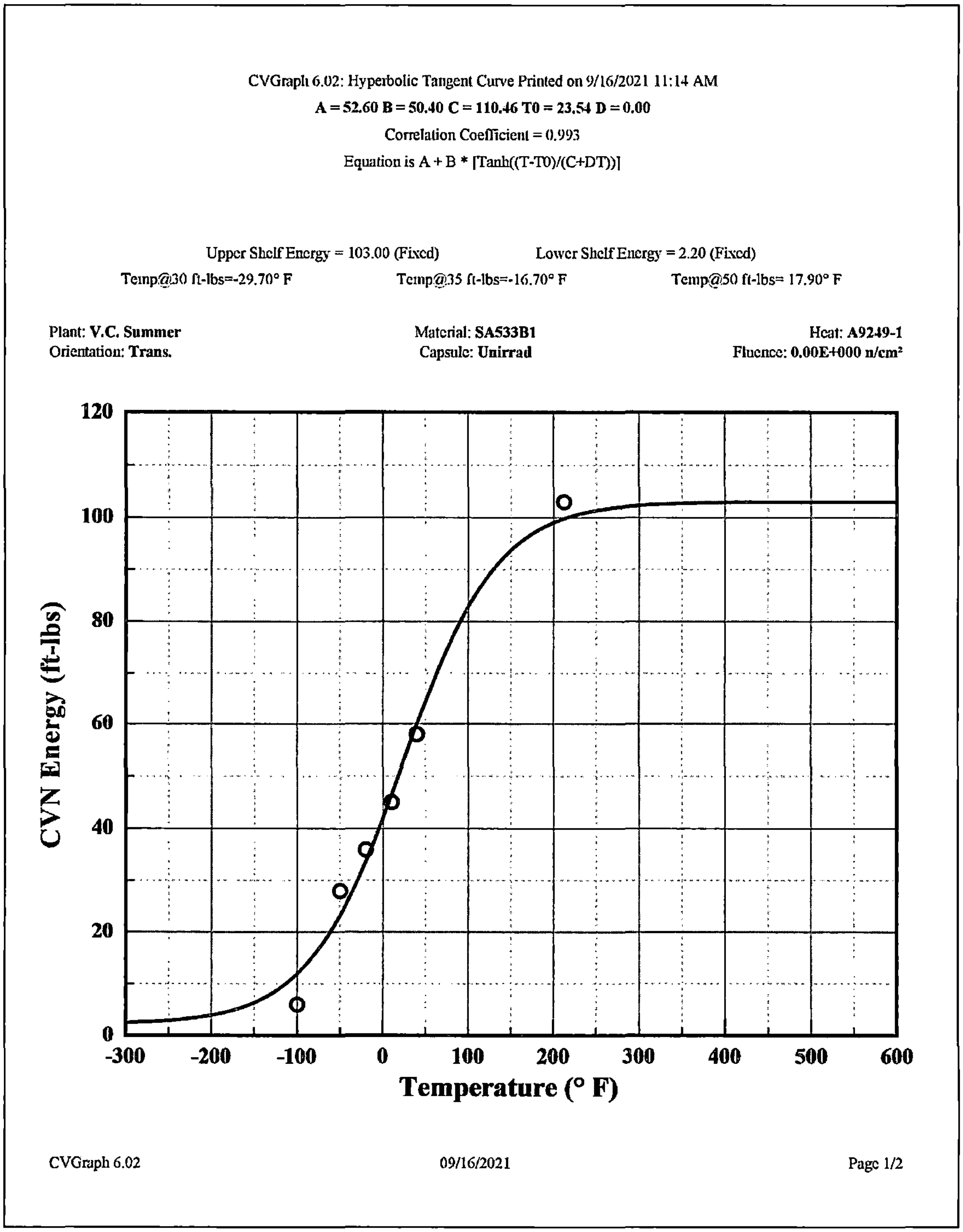


Figure B.15-1 Transition Ring Plates
Plot of Measured Transverse Direction CVN Data (cont.)

Plant: **V.C. Summer** Material: **SA533B1** Heat: **A9249-1**
Orientation: **Trans.** Capsule: **Unirrad** Fluence: **0.00E+000 n/cm²**

Charpy V-Notch Data

Temperature (° F)	Input CVN	Computed CVN	Differential
-100	6.0	11.9	-5.93
-50	28.0	23.3	4.74
-20	36.0	33.7	2.30
10	45.0	46.5	-1.45
40	58.0	60.1	-2.05
212	103.0	99.8	3.22

CVGraph 6.02 09/16/2021 Page 2/2

Figure B.15-2 Transition Ring Plates
Plot of Measured Transverse Direction Lateral Expansion Data

CVGraph 6.02: Hyperbolic Tangent Curve Printed on 9/16/2021 11:16 AM

A = 40.50 B = 39.50 C = 113.21 T0 = 4.65 D = 0.00

Correlation Coefficient = 0.976

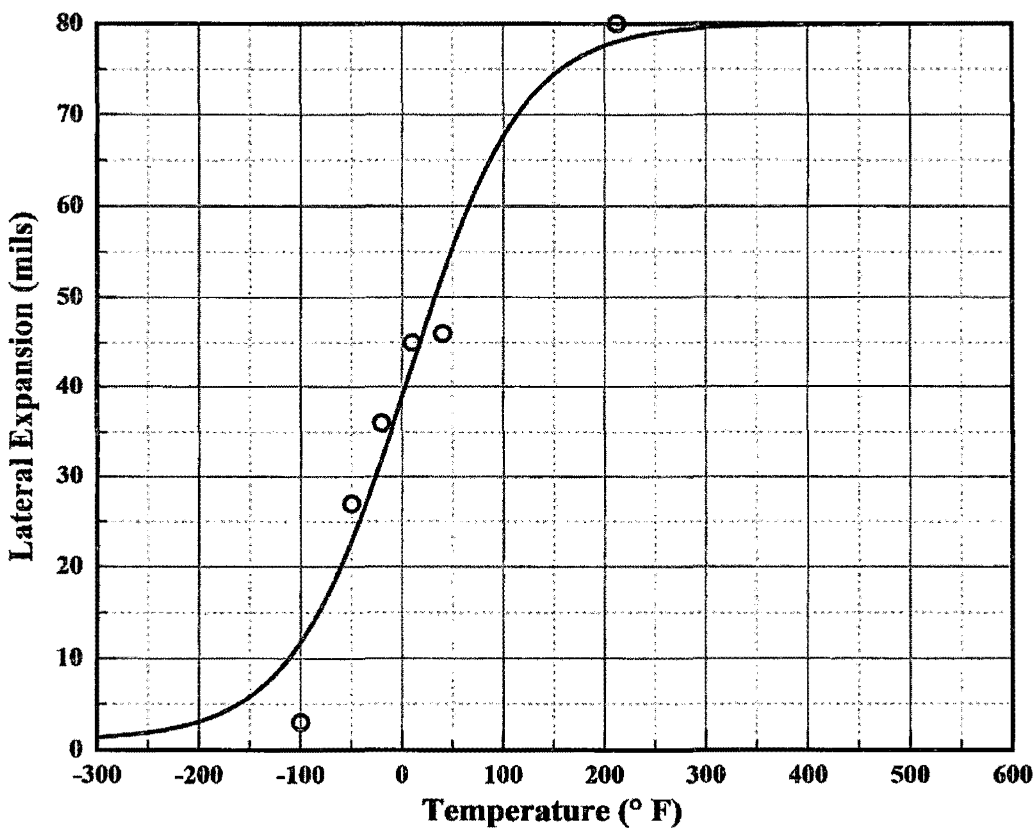
Equation is A + B * [Tanh((T-T0)/(C+DT))]

Upper Shelf L.E. = 80.00 (Fixed) Lower Shelf L.E. = 1.00 (Fixed)

Temp@35 mils=-11.20° F

Plant: **V.C. Summer** Material: **SA533B1** Heat: **A9249-1**

Orientation: **Trans.** Capsule: **Unirrad** Fluence: **0.00E+000 n/cm²**



CVGraph 6.02 09/16/2021 Page 1/2

Figure B.15-2 Transition Ring Plates
Plot of Measured Transverse Direction Lateral Expansion Data (cont.)

Plant: **V.C. Summer** Material: **SA533B1** Heat: **C9923-2**
Orientation: **Trans.** Capsule: **unirrad** Fluence: **0.00E+000 n/cm²**

Charpy V-Notch Data

Temperature (° F)	Input L. E.	Computed L. E.	Differential
-10	27.0	21.4	5.61
50	38.0	40.5	-2.50
40	34.0	37.1	-3.09
70	45.0	47.2	-2.17
120	65.0	60.9	4.07
212	76.0	72.6	3.37

CVGraph 6.02 09/15/2021 Page 2/2

B.16 V.C. Summer Unit 1 Bottom Head, Heat # A9231-2

Tables B.16-1 and B.16-2 summarize all available Charpy V-notch test data and drop-weight test data taken from the V.C. Summer Unit 1 CMTRs for the Bottom Head.

Table B.16-1 Charpy V-Notch Test Data for the Bottom Head, Heat # A9231-2

Longitudinal				Transverse			
Temp. (°F)	CVN Impact Energy (ft-lb)	Lateral Expansion (mils)	Shear (%)	Temp. (°F)	CVN Impact Energy (ft-lb)	Lateral Expansion (mils)	Shear (%)
-100	10	7	10	-100	5[a]	3	10
-100	6	4	10	-100	7	3	10
-100	6[a]	4[a]	10	-100	6	2[a]	10
-20	26	22	20	-20	42	23[a]	30
-20	17[a]	16[a]	20	-20	28[a]	35	30
-20	22	26	20	-20	47	34	30
10	26[a]	32	40	10	23[a]	33	40
10	38	24[a]	40	10	53	22[a]	40
10	44	32	40	10	38	41	40
40	34[a]	28[a]	50	40	50	40	50
40	72	57	50	40	58	36[a]	50
40	62	49	50	40	48[a]	44	50
100	86	50[a]	70	100	92[a]	66[a]	80
100	77	62	70	100	102	70	80
100	76[a]	64	70	100	96	75	80
212	130	83	99	212	133[a][b]	86[a][b]	99
212	120[a][b]	81[a][b]	99	212	136	89	99
212	126	84	99	212	133	90	99

Notes for Table B.16-1:

(a) Minimum value used in the CVGRAPH plots in accordance with ASME Code III Subarticle NB-2331 criteria.

(b) The value fixed as the upper shelf in CVGRAPH plots.

Table B.16-2 Drop-Weight Test Data for Bottom Head, Heat # A9231-2

Test Temperature (°F)	Drop-Weights[(a)]	T_{NDT} (°F)
10	1–NF	-10
0	2–NF	
-10	1–NF, 1–F	
-20	1–F	

Note for Table B.16-2:
(a) NF = "No Fail," F = "Fail".

B.16.1 Determination of the Initial RT_{NDT}

Using the data summarized in Tables B.16-1 and B.16-2, the initial RT_{NDT} value can be determined in accordance with the ASME Code Section III, Subarticle NB-2331 requirements. Following the requirements of ASME Code Section III, Subarticle NB-2331, Paragraph (a)(2), the minimum Charpy V-notch test data is first checked at a temperature not greater than the drop-weight T_{NDT} (or NDT) plus 60°F to determine if the material exhibits at least 50 ft-lb absorbed energy and 35 mils LE in the "weak" direction. Charpy V-notch tests were conducted at 40°F, which is less than T_{NDT} + 60°F (-10°F + 60°F = 50°F). The minimum Charpy V-notch test data at this temperature did **NOT** exhibit a minimum of 50 ft-lb absorbed energy and 35 mils lateral expansion; therefore, the Charpy V-notch tests at T_{NDT} + 60°F would **NOT** satisfy the criteria.

To precisely determine the temperature at which 50 ft-lb and 35 mils LE were obtained on the specimens, the unirradiated Charpy V-notch data may be plotted and fit using a hyperbolic tangent curve-fitting software, CVGRAPH. Only the minimum data points at each Charpy V-notch test temperature were used as input to the curve-fitting software, in accordance with ASME Code Section III, Subarticle NB-2331, Paragraph (a)(4). When plotting, the USE is fixed to the minimum Charpy impact energy or lateral expansion used in the plot which experience ≥ 95% shear. The resulting CVGRAPH figures are contained in the following pages for Charpy V-notch absorbed energy and lateral expansion.

Using these figures, the temperature at which 50 ft-lb absorbed energy and 35 mils lateral expansion were achieved may be obtained. The absorbed energy test data is more conservative than the lateral expansion test data; therefore, it becomes the dominant data set in defining initial RT_{NDT}. It is noted that while the longitudinal direction is usually considered the "strong" direction, some of the longitudinal data is more limiting than the transverse data. Therefore, both the longitudinal and transverse data will be fitted and the more limiting results will be used to determine the initial RT_{NDT}.

$$T_{50\ \text{ft-lb}} = \text{Max}\ [T_{50\ \text{ft-lb, long.}}, T_{50\ \text{ft-lb, trans.}}] = \text{Max}\ [59.9°\text{F}, 42.5°\text{F}] = 59.9°\text{F}$$
$$T_{35\ \text{mils}} = \text{Max}\ [T_{35\ \text{mils, long.}}, T_{35\ \text{mils, trans.}}] = \text{Max}\ [52.5°\text{F}, 32.5°\text{F}] == 52.5°\text{F}$$
$$T_{Cv} = \text{Max}\ [T_{50\ \text{ft-lb}}, T_{35\ \text{mils}}] = \text{Max}\ [59.9°\text{F}, 52.5°\text{F}]$$
$$T_{Cv} = 59.9°\text{F}$$

Following the requirements of ASME Code Section III, Subarticle NB-2331, Paragraph (a)(3), the initial RT_{NDT} is the higher of T_{NDT} (determined from the drop-weight tests) and T_{Cv} (determined above) minus 60°F.

RT_{NDT} = Max [T_{NDT}, T_{CV} - 60°F]
RT_{NDT} = Max [-10°F, 59.9°F - 60°F] = Max [-10°F, -0.1°F]

Bottom Head Initial RT_{NDT} = 0°F

B.16.2 Determination of the Initial USE

The current 10 CFR 50, Appendix G requirements specify that USE be calculated based on ASTM E185-82. Herein, USE is calculated based on an interpretation of ASTM E185-82 that is best explained by the most recent version of the ASTM E185 manual (2016 version). Using the guidelines in ASTM E185-82 and E185-16, the average of all Charpy data ≥ 95% shear is reported as the USE. In some instances, there may be data deemed 'out of family,' which are removed from the determination of the USE based on engineering judgment. However, the use of engineering judgment to remove 'out of family' data was not necessary for this material. The transverse (weak direction) USE is displayed below; this value is the average of each of the impact energy values contained in Table B.16-1 with shear ≥ 95%.

Bottom Head Initial USE = Average (133, 136, 133) ft-lb
= 134 ft-lb

However, the longitudinal Charpy data in Table B.16-1 exhibit lower impact energies at ≥ 95% shear than the axial data. Therefore, the USE will be recalculated using the longitudinal impact energy data with shear ≥ 95% because it represents the lower bound USE. It is noted the BTP 5-3 methodology is NOT being implemented here, which reduces the longitudinally oriented impact energies to 65% of the reported values in order to conservatively estimate transversely oriented specimens. This is because transverse data is available and does not need to be estimated.

Bottom Head Initial USE = Average (130, 120, 126) ft-lb
= 125 ft-lb

B.16.3 Chemistry

The Cu and Ni wt. % chemical compositions of the V.C. Summer Unit 1 reactor vessel materials were defined by a review of the available original test documentation. The material's chemical properties are defined as the average of all available data. When component specific data was not available, a generic value was defined as the mean plus one standard deviation of available data from similar materials. This method is consistent with Regulatory Guide 1.99, Revision 2, which allows the mean plus one standard deviation method to be used for conservative chemistry estimates based on generic data if component specific data is not available. The chemical compositions are summarized in Table B.16-3.

Table B.16-3 Chemistry Data for Bottom Head, Heat # A9231-2

Copper (wt.-%)	Nickel (wt.-%)	Source
-	0.45	CMTR, Lukens Steel Analysis
0.172	-	Generic value based on a mean plus one standard deviation analysis of the high copper A533, Grade B, Class 1, plate materials contained in Table G.2 of ORNL/TM-2006/530.

Therefore, the chemical content will be defined as shown below going forward:

Bottom Head Cu Content	=	**0.172 wt-%**
Bottom Head Ni Content	=	**0.45 wt-%**

Figure B.16-1 Bottom Head
Plot of Measured Transverse Direction CVN Data

CVGraph 6.02: Hyperbolic Tangent Curve Printed on 9/16/2021 11:22 AM

A = 67.60 B = 65.40 C = 86.70 T0 = 66.42 D = 0.00

Correlation Coefficient = 0.993

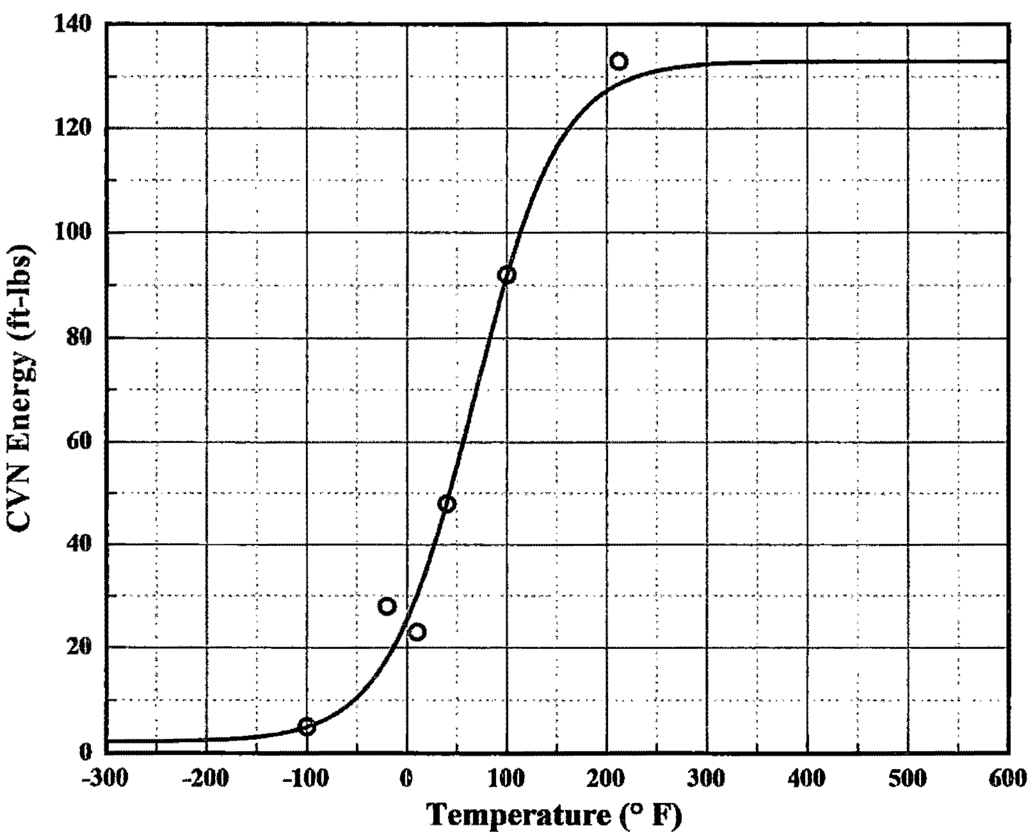
Equation is A + B * [Tanh((T-T0)/(C+DT))]

Upper Shelf Energy = 133.00 (Fixed) Lower Shelf Energy = 2.20 (Fixed)

Temp@30 ft-lbs= 9.70° F Temp@35 ft-lbs= 19.00° F Temp@50 ft-lbs= 42.50° F

Plant: **V.C. Summer** Material: **SA533B1** Heat: **A9231-2**

Orientation: **Trans.** Capsule: **Unirrad** Fluence: **0.00E+000 n/cm²**



CVGraph 6.02 09/16/2021 Page 1/2

Figure B.16-1 Bottom Head
Plot of Measured Transverse Direction CVN Data (cont.)

Plant: **V.C. Summer**
Orientation: **Trans.**

Material: **SA533B1**
Capsule: **Unirrad**

Heat: **A9231-2**
Fluence: **0.00E+000 n/cm²**

Charpy V-Notch Data

Temperature (° F)	Input CVN	Computed CVN	Differential
-100	5.0	5.0	0.05
-20	28.0	17.9	10.12
10	23.0	30.2	-7.18
40	48.0	48.3	-0.27
100	92.0	91.7	0.26
212	133.0	128.6	4.40

CVGraph 6.02 09/16/2021 Page 2/2

Figure B.16-2 Bottom Head
Plot of Measured Transverse Direction Lateral Expansion Data

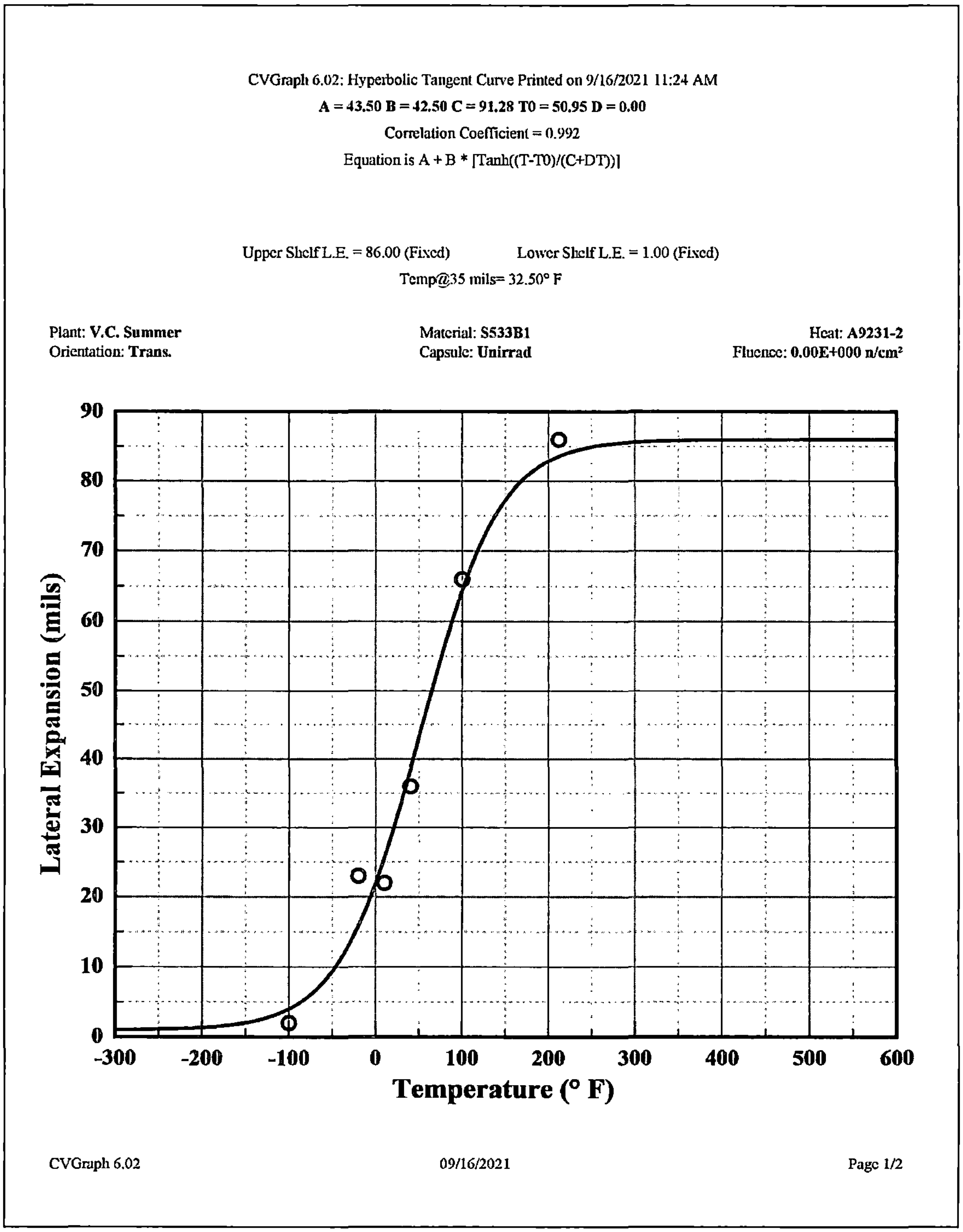


Figure B.16-2 Bottom Head
Plot of Measured Transverse Direction Lateral Expansion Data (cont.)

Plant: **V.C. Summer**
Orientation: **Trans.**

Material: **S533B1**
Capsule: **Unirrad**

Heat: **A9231-2**
Fluence: **0.00E+000 n/cm²**

Charpy V-Notch Data

Temperature (° F)	Input L. E.	Computed L. E.	Differential
-100	2.0	4.0	-2.00
-20	23.0	15.8	7.17
10	22.0	25.6	-3.62
40	36.0	38.4	-2.43
100	66.0	64.4	1.63
212	86.0	83.6	2.42

CVGraph 6.02 09/16/2021 Page 2/2

Figure B.16-3 Bottom Head
Plot of Measured Longitudinal Direction CVN Data

CVGraph 6.02: Hyperbolic Tangent Curve Printed on 9/16/2021 11:27 AM

A = 61.10 B = 58.90 C = 87.90 T0 = 76.66 D = 0.00

Correlation Coefficient = 0.997

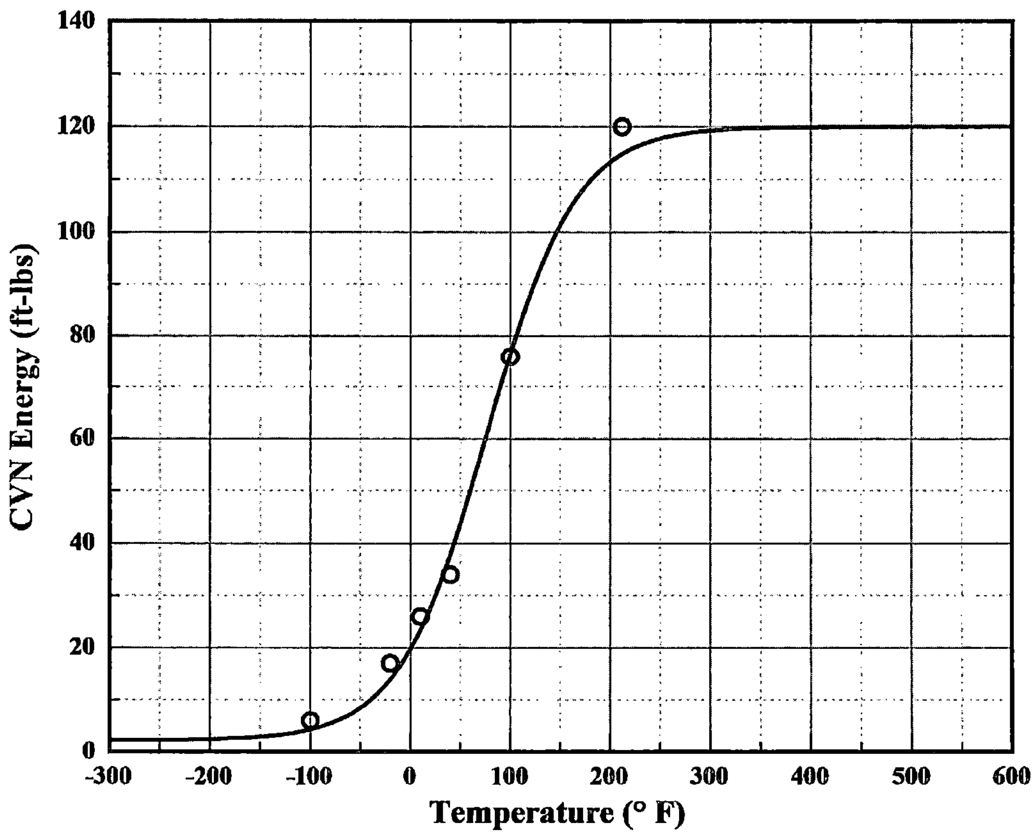
Equation is A + B * [Tanh((T-T0)/(C+DT))]

Upper Shelf Energy = 120.00 (Fixed) Lower Shelf Energy = 2.20 (Fixed)

Temp@30 ft-lbs= 25.10° F Temp@35 ft-lbs= 34.90° F Temp@50 ft-lbs= 59.90° F

Plant: **V.C. Summer** Material: **SA533B1** Heat: **A9231-2**

Orientation: **Long.** Capsule: **Unirrad** Fluence: **0.00E+000 n/cm²**



CVGraph 6.02 09/16/2021 Page 1/2

Figure B.16-3 Bottom Head
Plot of Measured Longitudinal Direction CVN Data (cont.)

Plant: **V.C. Summer**
Orientation: **Long.**

Material: **SA533B1**
Capsule: **Unirrad**

Heat: **A9231-2**
Fluence: **0.00E+000 n/cm²**

Charpy V-Notch Data

Temperature (° F)	Input CVN	Computed CVN	Differential
-100	6.0	4.3	1.72
-20	17.0	14.0	3.04
10	26.0	23.4	2.60
40	34.0	37.9	-3.87
100	76.0	76.4	-0.38
212	120.0	114.8	5.18

CVGraph 6.02 09/16/2021 Page 2/2

Figure B.16-4 Bottom Head
Plot of Measured Longitudinal Direction Lateral Expansion Data

CVGraph 6.02: Hyperbolic Tangent Curve Printed on 9/16/2021 11:26 AM

A = 41.00 B = 40.00 C = 108.90 T0 = 68.90 D = 0.00

Correlation Coefficient = 0.994

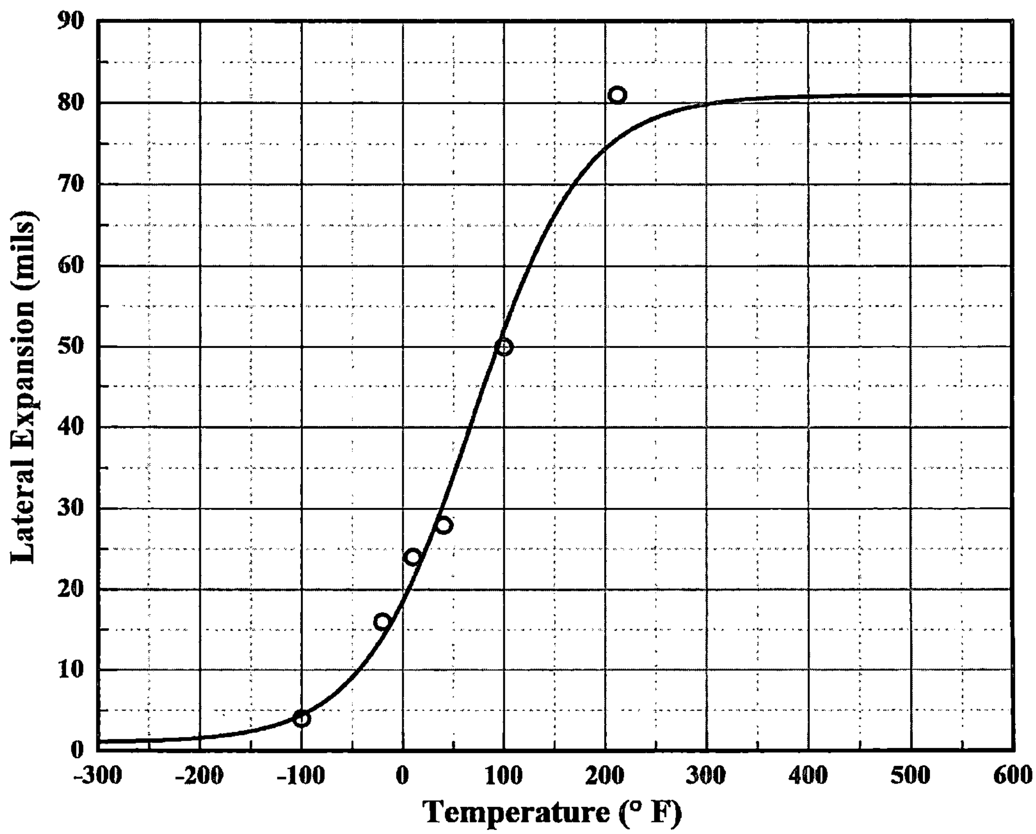
Equation is A + B * [Tanh((T-T0)/(C+DT))]

Upper Shelf L.E. = 81.00 (Fixed) Lower Shelf L.E. = 1.00 (Fixed)

Temp@35 mils= 52.50° F

Plant: **V.C. Summer** Material: **SA533B1** Heat: **A9231-2**

Orientation: **Long.** Capsule: **Unirrad** Fluence: **0.00E+000 n/cm²**



CVGraph 6.02 09/16/2021 Page 1/2

Figure B.16-4 Bottom Head
Plot of Measured Longitudinal Direction Lateral Expansion Data (cont.)

Plant: **V.C. Summer**
Orientation: **Long.**

Material: **SA533B1**
Capsule: **Unirrad**

Heat: **A9231-2**
Fluence: **0.00E+000 n/cm²**

Charpy V-Notch Data

Temperature (° F)	Input L. E.	Computed L. E.	Differential
-100	4.0	4.4	-0.44
-20	16.0	14.1	1.92
10	24.0	21.3	2.75
40	28.0	30.6	-2.63
100	50.0	52.1	-2.12
212	81.0	75.6	5.39

CVGraph 6.02 09/16/2021 Page 2/2

Attachment C
PA-MSC-1367, Tasks 1 – 3 Evaluations for V.C. Summer Unit 1 Reactor Vessel Welds

C.1 Introduction

This section evaluates material properties, i.e., Cu, Ni, unirradiated RT_{NDT}, & unirradiated USE, for all welds used in the fabrication of the V.C. Summer Unit 1 reactor vessel. These welds were identified from a review of Certified Material Test Reports (CMTRs), surveillance capsule program, and Chicago Bridge & Iron (CB&I) fabrication records available to Westinghouse. The weld heats identified in the fabrication records were cross-referenced against an investigation performed on CB&I fabricated vessels to address NRC IE Bulletin 78-12 regarding atypical weld metal in reactor pressure vessel welds. In some situations, in the presumed extended beltline, the heat number used in specific weld seams could not be identified. To address these situations, bounding or generic V.C. Summer weld properties will be developed to be used anywhere the specific weld heat cannot be identified.

C.2 Summary of Welds at V.C. Summer Unit 1

Two types of welds that were used in the fabrication of CB&I vessels, shielded metal arc welds (SMAWs) and submerged arc welds (SAWs). Where the geometry permitted automatic SAW was used and SMAW was applied where it was required to complete the welds due to geometry and for repairs. Each weld type is addressed separately in the following sections. Table C.2-1 summarizes all material properties of the SAW taken from the V.C. Summer Unit 1 reactor vessel fabrication files. Table C.2-2 summarizes all material properties of the SMAW taken from the V.C. Summer Unit 1 reactor vessel fabrication files. The more limiting will be used to generate the bounding/generic material properties.

The sections below discuss the development of each bounding/generic material properties.

C.2.1 Determination of the Initial RT_{NDT}

An initial RT_{NDT} has been established for all weld heats used in the V.C. Summer reactor vessel. Because this is a critical property for demonstrating the safe operation of the plant, it was determined that the maximum initial RT_{NDT} should be used in instances when the heat could not be determined.

The bounding RT_{NDT} for SAW is 10°F. Since this is a bounding value based on measured data, the standard deviation associated with its initial measurement is zero (σ_I = 0°F). This value is considered conservative as it is based on BTP 5-3, Position 1.1(4) which uses data at only a single temperature. It is also more conservative than the generic initial RT_{NDT} = -56°F for Linde 124 flux with a σ_I = 17°F permitted by 10 CFR 50.61, i.e., initial $RT_{NDT} + 2\sigma_I$ = -22°F.

The bounding RT_{NDT} for SMAW is 0°F. Since this is a bounding value based on measured data, the standard deviation associated with its initial measurement is zero (σ_I = 0°F). This value is considered conservative as it is based on BTP 5-3, Position 1.1(4) which uses data at only a single temperature. Instead, the T_{NDT} is assumed to be equal to the next test temperature, i.e., 0°F.

The bounding RT_{NDT} value between SAW and SMAW is 10°F.

Bounding Weld Initial RT_{NDT} = 10°F

C.2.2 Determination of the Initial USE

Roughly half the weld heats at V.C. Summer Unit 1 for both SAW and SMAW have shear data ≥ 95% required by ASTM E185 to establish USE. Herein, generic USE values are determined based on the mean USE of common weld types minus 2 standard deviations (σ). The mean USE is based on a review of all Charpy impact energy with shear data ≥ 95%. If insufficient data is available to determine the USE, such as all data indicates a shear less than 95%, an approximate USE based on the data available is reported as greater than the CVN impact energy with the maximum shear. If the highest shear includes multiple CVN impact energies, then the USE is set to greater than the largest impact energy. This USE value does not provide an accurate representation of USE; therefore, this data point is excluded from the statistical analysis.

The results indicate that the average SAW USE minus two standard deviations is 81 ft-lb. This value is considered conservative because most of the weld heats with shear data < 95% experienced CVN impact energies greater than 81 ft-lb prior to the onset of the USE. The exception, Heat # 3P4955, Flux Type Linde 124, Lot #s 1214 & 3478, experienced nearly 81 ft-lb (77 ft-lb & 67 ft-lb, respectively) with a shear value much less than 95% (75% & 65% shear, respectively).

The results indicate that the average SMAW USE minus two standard deviations is 80 ft-lb. This value is considered conservative since the majority of the weld heats with shear data < 95% experience CVN impact energies greater than 80 ft-lb prior to the onset of the USE. Those that do not experience > 80 ft-lb also have shears much less than 95%. For example, Heat # 04P046, Lot # D217A27A experience only 40 ft-lb, but also experience only 30% shear.

The bounding generic USE value between SAW and SMAW is subsequently 80 ft-lb. This is greater than the 10 CFR 50, Appendix G minimum unirradiated USE value (75 ft-lb) and the irradiated USE screening criterion for operating plants (50 ft-lb).

Generic Weld Initial USE = 80 ft-lb

C.2.3 Chemistry

This value is based on the Regulatory Guide 1.99, Revision 2 mean plus one standard deviation approach. The generic weight percent values of 0.06% for Cu and 1.01% for Ni can be utilized for V.C. Summer reactor vessel SAWs and/or SMAW when insufficient data is available to determine weld-specific chemistry values.

Generic Weld Cu Content = 0.06 wt.-%

Generic Weld Ni Content = 1.01 wt.-%

Table C.2-1 Summary of Material Properties for SAWs[(a)]

Heat #	Flux Type	Lot #	Cu (wt-%)	Ni (wt-%)	Initial RT_{NDT}		USE[(c)] (ft-lb)
					(°F)	Method[(b)]	
4P4784	Linde 124	3930	0.05	0.91	-49	ASME	86
3P4955	Linde 124	1214	0.03	0.98	-20	ASME	> 77 ft-lb @ 75% Shear
3P4955	Linde 124	3478	0.03	0.97	-20	ASME	> 67 ft-lb @ 70% Shear
3P4966	Linde 124	1214	0.03	0.90	-20	ASME	> 88 ft-lb @ 85% Shear
3P4966	Linde 124	0331	0.03	1.07	10	BTP 5-3	> 95 ft-lb @ 70% Shear
5P5657	Linde 124	0931	0.07	0.89	-60	ASME	88
5P6214B	Linde 124	0331	0.02	0.82	-40	ASME	90
5P6771	Linde 124	0342	0.03	0.94	-20	ASME	83
Maximum			0.07	1.07	10		> 95 ft-lb @ 70% Shear
Minimum			0.02	0.82	-60		> 67 ft-lb @ 70% Shear
Average			0.04	0.94	-		87
Standard Deviation (σ)			0.02	0.07	-		3
Average ± σ			0.06	1.01	-		84
Average ± 2σ			-	-	-		81

Notes for Table C.2-1:

(a) The material properties are generated in subsequent sections of this attachment.

(b) "ASME" indicates the initial RT_{NDT} value was developed be on ASME NB-2300 methods. "BTP 5-3" indicates that BTP 5-3, Position 1.1(4) was utilized because of lack of sufficient testing data to satisfy the ASME NB-2300 methods requirements.

(c) USE values preceded by a greater than or equal to symbol, ">", identifies a material with no shear data ≥ 95%; thus, the initial USE values for these materials were set to greater than the impact energy with highest shear. The percent value identifies the shear value corresponding to the lower bound USE. These data points are excluded from the statistical analysis.

Table C.2-2 Summary of Material Properties for SMAWs[(a)]

Heat #	Lot #	Cu (wt-%)	Ni (wt-%)	Initial RT_{NDT} (°F)	Initial RT_{NDT} Method[(b)]	USE[(c)] (ft-lb)
492L4871	A421B27AF	0.03	0.98	-60	ASME	130
492L4871	A421B27AE	0.04	0.95	-60	ASME	157
627069	C312A27AG	0.02	0.99	-60	ASME	115
624039	D205A27A	0.06	0.92	-77	ASME	119
422K8511	G313A27AD	0.01	1.00	-78	ASME	142
627184	C314A27AH	0.03	1.01	-50	ASME	102
626677	C301A27AF	0.02	0.95	-20	ASME	93
05T776	L314A27AH	0.06	0.92	-50	ASME	119
624263	E204A27A	0.06	0.89	-20	ASME	> 73.5 @ 75%
421A6811	F022A27A	0.07	0.88	-20	ASME	> 91 @ 75%
07L669	K004A27A	0.03	1.02	-20	BTP 5-3	> 50 @ 60%
C3L46C	J020A27A	0.02	0.87	-20	ASME	> 57 @ 65%
422B7201	L030A27A	0.04	0.90	-20	ASME	> 66 @ 70%
09L853	A111A27A	0.03	0.86	-20	BTP 5-3	> 78 @ 80%
08M365	G128A27A	0.02	1.10	-20	BTP 5-3	> 51 @ 60%
421E0601	L117A27A	0.03	0.87	-20	ASME	> 102.5 @ 90%
09M814	L115A27A	0.03	0.92	-20	ASME	> 88 @ 80%
09M814	L114A27A	0.01	0.82	-20	ASME	> 92 @ 80%
623275	L121A27A	0.05	0.84	-20	ASME	> 80 @ 75%
627260	B322A27AE	0.06	1.08	-20	ASME	> 80 @ 25%
624039	D224A27A	0.07	1.01	-20	ASME	> 86 @ 63%
05P018	D211A27A	0.09	0.90	-20	ASME	> 85 @ 70%
421Z0611	L908A27A	0.03	1.01	-20	ASME	> 113.5 @ 85%
04P046	D217A27A	0.06	0.90	0	BTP 5-3	> 40 @ 30%
624063	C228A27A	0.03	1.00	-20	ASME	> 70 @ 60%
Maximum		0.09	1.10	0		157
Minimum		0.01	0.82	-78		> 40 @ 30%
Average		0.04	0.94	-31		122
Standard Deviation (σ)		0.02	0.07	-		21
Average ± σ		0.06	1.01	-		101
Average ± 2σ		-	-	-		80

Notes for Table C.2-1:

(a) The material properties are generated in subsequent sections of this attachment.

(b) "ASME" indicates the initial RT_{NDT} value was developed be on ASME NB-2300 methods. "BTP 5-3" indicates that BTP 5-3, Position 1.1(4) was utilized because of lack of sufficient testing data to satisfy the ASME NB-2300 methods requirements.

(c) USE values preceded by a greater than or equal to symbol, ">", identifies a material with no shear data ≥ 95%; thus, the initial USE values for these materials were set to greater than the impact energy with highest shear. The percent value identifies the shear value corresponding to the lower bound USE. These data points are excluded from the statistical analysis.

C.3 V.C. Summer Unit 1, Heat # 3P4955, Linde 124 Flux, Lot # 1214

Tables C.3-1 and C.3-2 summarize all available Charpy V-notch test data and drop-weight test data taken from the V.C. Summer Unit 1 reactor vessel fabrication files. The material properties were evaluated in two ways. 1st, the material properties were evaluated with the weld being deposited as a single wire. 2nd, the material properties were evaluated with the weld being deposited by tandem wires. For the purpose of this analysis, both will be evaluated, and the limiting property will be used.

Table C.3-1 Charpy V-Notch Test Data for the Weld Heat # 3P4955, Linde 124 Flux, Lot # 1214

Temp. (°F)	CVN Impact Energy (ft-lb)	Lateral Expansion (mils)	Shear (%)
Single Wire			
10	45	38	25
10	36	34	25
10	52	50	30
10	55	48	30
10	52	46	30
40	62	51	60
40	77	64	75
40	59	53	50
Tandem Wire			
10	69	64	50
10	67	53	50
10	67	58	50
10	61	54	50
10	64	67	50
40	52	45	35
40	63	53	45
40	53	44	30

Table C.3-2 Drop-Weight Test Data for Weld Heat # 3P4955, Linde 124 Flux, Lot # 1214

	Test Temperature (°F)	Drop-Weights[(a)]	T_{NDT} (°F)
Single	-10	1–NF	-20[(b)]
	-10	1–NF	
Tandem	-10	1–NF	-20[(b)]
	-10	1–NF	

Note for Table C.3-2:

(a) NF = "No Fail," F = "Fail".

(b) Drop-weight testing had no breaks at the lowest test temperature, i.e., -10°F; therefore, the NDT less than or equal to the next test temperature, i.e., -20°F.

C.3.1 Determination of the Initial RT_{NDT}

Using the data summarized in Tables C.3-1 and C.3-2, the initial RT_{NDT} value can be determined in accordance with the ASME Code Section III, Subarticle NB-2331 requirements. Following the requirements of ASME Code Section III, Subarticle NB-2331, Paragraph (a)(2), the minimum Charpy V-notch test data is first checked at a temperature not greater than the drop-weight T_{NDT} (or NDT) plus 60°F to determine if the material exhibits at least 50 ft-lb absorbed energy and 35 mils LE.

For both the single and tandem wire analysis, the Charpy V-notch tests were conducted at 40°F, T_{NDT} + 60°F (-20°F + 60°F = 40°F). The minimum Charpy V-notch test data at this temperature exhibit a minimum of 50 ft-lb absorbed energy and 35 mils lateral expansion; therefore, the Charpy V-notch tests at T_{NDT} + 60°F satisfy the criteria. Per ASME Code Section III, Subarticle NB-2331, Paragraph (a)(2), the requirements have been met such that T_{NDT} is the initial reference temperature RT_{NDT}.

RT_{NDT} = -20°F

Weld Heat # 3P4955, Linde 124 Flux, Lot # 1214 Initial RT_{NDT} = -20°F

C.3.2 Determination of the Initial USE

The current 10 CFR 50, Appendix G requirements specify that USE be calculated based on ASTM E185-82. Herein, USE is calculated based on an interpretation of ASTM E185-82 that is best explained by the most recent version of the ASTM E185 manual (2016 version). Using the guidelines in ASTM E185-82 and E185-16, the average of all Charpy data ≥ 95% shear is reported as the USE. In some instances, there may be data deemed 'out of family,' which are removed from the determination of the USE based on engineering judgment. However, the use of engineering judgment to remove 'out of family' data was not necessary for this material. Per Table C.3-1, no data with known shear ≥ 95% exists. Thus, the USE is set to greater than the impact energy with highest shear. If the highest shear includes multiple CVN impact energies, then the USE is set to greater than the largest impact energy. The USE for the material will be defined as the USE value with highest shear between the single and tandem wire.

Single Wire, Weld Heat # 3P4955, Linde 124 Flux, Lot # 1214 Initial USE > 77 ft-lb @ 75% Shear

Tandem Wire, Weld Heat # 3P4955, Linde 124 Flux, Lot # 1214 Initial USE > 69 ft-lb @ 50% Shear

Weld Heat # 3P4955, Linde 124 Flux, Lot # 1214 Initial USE > 77 ft-lb @ 75% Shear

C.3.3 Chemistry

The Cu and Ni wt. % chemical compositions of the V.C. Summer Unit 1 reactor vessel materials were defined by a review of the available original test documentation. If data for single or tandem wire includes multiple measurements, then the average of all available data will be used. The limiting value between the single and tandem weld will be used. The chemical compositions are summarized in Table C.3-3.

Table C.3-3 Chemistry Data for Weld Heat # 3P4955, Linde 124 Flux, Lot # 1214

Copper (wt.-%)	**Nickel (wt.-%)**	**Source**
0.025	0.94	Single Wire Check Analysis
0.022	0.98	Tandem Wire Check Analysis

Therefore, to determine the generic chemical content of the welds in the V.C. Summer reactor vessel, the below values will be used:

Weld Heat # 3P4955, Linde 124 Flux, Lot # 1214 Cu Content = 0.03 wt-%

Weld Heat # 3P4955, Linde 124 Flux, Lot # 1214 Ni Content = 0.98 wt-%

C.4 V.C. Summer Unit 1, Heat # 3P4955, Linde 124 Flux, Lot # 3478

Tables C.4-1 and C.4-2 summarize all available Charpy V-notch test data and drop-weight test data taken from the V.C. Summer Unit 1 vessel fabrication files. The material properties were evaluated in two ways. 1st, the material properties were evaluated with the weld being deposited as a single wire. 2nd, the material properties were evaluated with the weld being deposited by tandem wires. For the purpose of this analysis, both will be evaluated, and the limiting property will be used.

Table C.4-1 Charpy V-Notch Test Data for the Weld Heat # 3P4955, Linde 124 Flux, Lot # 3478

Temp. (°F)	CVN Impact Energy (ft-lb)	Lateral Expansion (mils)	Shear (%)
Single Wire			
10	44.5	45	45
10	55.5	50	55
10	52.5	55	55
10	53.5	55	55
10	32.5	40	39
40	65	61	60
40	58	56	50
40	54.5	54	50
Tandem Wire			
10	67	66	70
10	65.5	69	70
10	59	59	70
10	48.5	46	50
10	51.5	53	55
40	56	49	55
40	56	56	50
40	60	46	50

Table C.4-2 Drop-Weight Test Data for Weld Heat # 3P4955, Linde 124 Flux, Lot # 3478

	Test Temperature (°F)	Drop-Weights[(a)]	T_{NDT} (°F)
Single	-10	1–NF	-20[(b)]
	-10	1–NF	
Tandem	-10	1–NF	-20[(b)]
	-10	1–NF	

Note for Table C.4-2:

(a) NF = "No Fail," F = "Fail".

(b) Drop-weight testing had no breaks at the lowest test temperature, i.e., -10°F; therefore, the NDT less than or equal to the next test temperature, i.e., -20°F.

C.4.1 Determination of the Initial RT_{NDT}

Using the data summarized in Tables C.4-1 and C.4-2, the initial RT_{NDT} value can be determined in accordance with the ASME Code Section III, Subarticle NB-2331 requirements. Following the requirements of ASME Code Section III, Subarticle NB-2331, Paragraph (a)(2), the minimum Charpy V-notch test data is first checked at a temperature not greater than the drop-weight T_{NDT} (or NDT) plus 60°F to determine if the material exhibits at least 50 ft-lb absorbed energy and 35 mils LE.

For both the single and tandem wires, the Charpy V-notch tests were conducted at 40°F, T_{NDT} + 60°F (-20°F + 60°F = 40°F). The minimum Charpy V-notch test data at this temperature exhibit a minimum of 50 ft-lb absorbed energy and 35 mils lateral expansion; therefore, the Charpy V-notch tests at T_{NDT} + 60°F satisfy the criteria. Per ASME Code Section III, Subarticle NB-2331, Paragraph (a)(2), the requirements have been met such that T_{NDT} is the initial reference temperature RT_{NDT}.

RT_{NDT} = -20°F

Weld Heat # 3P4955, Linde 124 Flux, Lot # 3478 Initial RT_{NDT} = -20°F

C.4.2 Determination of the Initial USE

The current 10 CFR 50, Appendix G requirements specify that USE be calculated based on ASTM E185-82. Herein, USE is calculated based on an interpretation of ASTM E185-82 that is best explained by the most recent version of the ASTM E185 manual (2016 version). Using the guidelines in ASTM E185-82 and E185-16, the average of all Charpy data ≥ 95% shear is reported as the USE. In some instances, there may be data deemed 'out of family,' which are removed from the determination of the USE based on engineering judgment. However, the use of engineering judgment to remove 'out of family' data was not necessary for this material. Per Table C.4-1, no data with known shear ≥ 95% exists. Thus, the USE is set to greater than the impact energy with highest shear. If the highest shear includes multiple CVN impact energies, then the USE is set to greater than the largest impact energy. The USE for the material will be defined as the USE value with highest shear between the single and tandem wire.

Single Wire, Weld Heat # 3P4955, Linde 124 Flux, Lot # 3478 Initial USE > 65 ft-lb @ 60% Shear

Tandem Wire, Weld Heat # 3P4955, Linde 124 Flux, Lot # 3478 Initial USE > 67 ft-lb @ 70% Shear

Weld Heat # 3P4955, Linde 124 Flux, Lot # 3478 Initial USE > 67 ft-lb @ 70% Shear

C.4.3 Chemistry

The Cu and Ni wt. % chemical compositions of the V.C. Summer Unit 1 reactor vessel materials were defined by a review of the available original test documentation. If data for single or tandem wire includes multiple measurements, then the average of all available data will be used. The limiting value between the single and tandem weld will be used. The chemical compositions are summarized in Table C.4-3.

Table C.4-3 Chemistry Data for Weld Heat # 3P4955, Linde 124 Flux, Lot # 3478

Copper (wt.-%)	Nickel (wt.-%)	Source
0.025	0.97	Single Wire Check Analysis
0.025	0.95	Tandem Wire Check Analysis

Therefore, to determine the generic chemical content of the welds in the V.C. Summer reactor vessel, the below values will be used:

Weld Heat # 3P4955, Linde 124 Flux, Lot # 3478 Cu Content = 0.03 wt-%

Weld Heat # 3P4955, Linde 124 Flux, Lot # 3478 Ni Content = 0.97 wt-%

C.5 V.C. Summer Unit 1, Heat # 3P4966, Linde 124 Flux, Lot # 0331

Tables C.5-1 summarizes all available Charpy V-notch test data taken from the V.C. Summer Unit 1 reactor vessel fabrication files. The material properties were evaluated in two ways. 1st, the material properties were evaluated with the weld being deposited as a single wire. 2nd, the material properties were evaluated with the weld being deposited by tandem wires. For the purpose of this analysis, both will be evaluated, and the limiting property will be used. No drop-weight test data is available.

Table C.5-1 Charpy V-Notch Test Data for the Weld Heat # 3P4966, Linde 124 Flux, Lot # 0331

Temp. (°F)	CVN Impact Energy (ft-lb)	Lateral Expansion (mils)	Shear (%)
Single Wire			
10	54	49	25
10	48	40	25
10	70	54	35
10	74	66	50
10	64	58	40
Tandem Wire			
10	77	65	50
10	90	66	70
10	80	59	35
10	95	71	70
10	78	62	35

C.5.1 Determination of the Initial RT_{NDT}

Since there in no drop-weight data and tests were performed at a single temperature, NUREG-0800 BTP 5-3 Position 1.1(4) may be used as an estimate of RT_{NDT}. Per Table C.5-1, all impact energies are greater than 45 ft-lb at 10°F; therefore, the initial RT_{NDT} is:

Weld Heat # 3P4966, Linde 124 Flux, Lot # 0331 Initial RT_{NDT} = 10°F

C.5.2 Determination of the Initial USE

The current 10 CFR 50, Appendix G requirements specify that USE be calculated based on ASTM E185-82. Herein, USE is calculated based on an interpretation of ASTM E185-82 that is best explained by the most recent version of the ASTM E185 manual (2016 version). Using the guidelines in ASTM E185-82 and E185-16, the average of all Charpy data ≥ 95% shear is reported as the USE. In some instances, there may be data deemed 'out of family,' which are removed from the determination of the USE based on engineering judgment. However, the use of engineering judgment to remove 'out of family' data was not necessary for this material. Per Table C.5-1, no data with known shear ≥ 95% exists. Thus, the USE is set to greater than the impact energy with highest shear. If the highest shear includes multiple CVN impact energies, then the

USE is set to greater than the largest impact energy. The USE for the material will be defined as the USE value with highest shear between the single and tandem wire.

Single Wire, Weld Heat # 3P4966, Linde 124 Flux, Lot # 0331 Initial USE > 74 ft-lb @ 50% Shear

Tandem Wire, Weld Heat # 3P4966, Linde 124 Flux, Lot # 0331 Initial USE > 95 ft-lb @ 70% Shear

Weld Heat # 3P4966, Linde 124 Flux, Lot # 0331 Initial USE > 95 ft-lb @ 70% Shear

C.5.3 Chemistry

The Cu and Ni wt. % chemical compositions of the V.C. Summer Unit 1 reactor vessel materials were defined by a review of the available original test documentation. If data for single or tandem wire includes multiple measurements, then the average of all available data will be used. The limiting value between the single and tandem weld will be used. The chemical compositions are summarized in Table C.5-2.

Table C.5-2 Chemistry Data for Weld Heat # 3P4966, Linde 124 Flux, Lot # 0331

Copper (wt.-%)	Nickel (wt.-%)	Source
0.023	1.07	Single Wire Check Analysis
0.025	0.94	Tandem Wire Check Analysis

Therefore, to determine the generic chemical content of the welds in the V.C. Summer reactor vessel, the below values will be used:

Weld Heat # 3P4966, Linde 124 Flux, Lot # 0331 Cu Content = 0.03 wt-%

Weld Heat # 3P4966, Linde 124 Flux, Lot # 0331 Ni Content = 1.07 wt-%

C.6 V.C. Summer Unit 1, Heat # 3P4966, Linde 124 Flux, Lot # 1214

Tables C.6-1 and C.6-2 summarize all available Charpy V-notch test data and drop-weight test data taken from the V.C. Summer Unit 1 reactor vessel fabrication files. The material properties were evaluated in two ways. 1st, the material properties were evaluated with the weld being deposited as a single wire. 2nd, the material properties were evaluated with the weld being deposited by tandem wires. For the purpose of this analysis, both will be evaluated, and the limiting property will be used.

Table C.6-1 Charpy V-Notch Test Data for the Weld Heat # 3P4966, Linde 124 Flux, Lot # 1214

Temp. (°F)	CVN Impact Energy (ft-lb)	Lateral Expansion (mils)	Shear (%)
Single Wire			
10	39	68	70
10	38	64	60
10	38	63	70
10	82	81	80
10	84	72	70
40	88	72	75
40	88	68	85
40	82	67	65
Tandem Wire			
10	28	18	30
10	84	62	40
10	63	57	40
10	75	51	40
10	78	57	40
40	65	52	55
40	69	55	65
40	70	59	45

Table C.6-2 Drop-Weight Test Data for Weld Heat # 3P4966, Linde 124 Flux, Lot # 1214

	Test Temperature (°F)	Drop-Weights[(a)]	T_{NDT} (°F)
Single	-10	1–NF	-20[(b)]
	-10	1–NF	
Tandem	-10	1–NF	-20[(b)]
	-10	1–NF	

Note for Table C.6-2:

(a) NF = "No Fail," F = "Fail".

(b) Drop-weight testing had no breaks at the lowest test temperature, i.e., 0°F; therefore, the NDT ≤ the next test temperature, i.e., -10°F.

C.6.1 Determination of the Initial RT_{NDT}

Using the data summarized in Tables C.6-1 and C.6-2, the initial RT_{NDT} value can be determined in accordance with the ASME Code Section III, Subarticle NB-2331 requirements. Following the requirements of ASME Code Section III, Subarticle NB-2331, Paragraph (a)(2), the minimum Charpy V-notch test data is first checked at a temperature not greater than the drop-weight T_{NDT} (or NDT) plus 60°F to determine if the material exhibits at least 50 ft-lb absorbed energy and 35 mils LE.

For both the single and tandem wire analysis, the Charpy V-notch tests were conducted at 40°F, T_{NDT} + 60°F (-20°F + 60°F = 40°F). The minimum Charpy V-notch test data at this temperature exhibit a minimum of 50 ft-lb absorbed energy and 35 mils lateral expansion; therefore, the Charpy V-notch tests at T_{NDT} + 60°F satisfy the criteria. Per ASME Code Section III, Subarticle NB-2331, Paragraph (a)(2), the requirements have been met such that T_{NDT} is the initial reference temperature RT_{NDT}.

RT_{NDT} = -20°F

Weld Heat # 3P4966, Linde 124 Flux, Lot # 1214 Initial RT_{NDT} = -20°F

C.6.2 Determination of the Initial USE

The current 10 CFR 50, Appendix G requirements specify that USE be calculated based on ASTM E185-82. Herein, USE is calculated based on an interpretation of ASTM E185-82 that is best explained by the most recent version of the ASTM E185 manual (2016 version). Using the guidelines in ASTM E185-82 and E185-16, the average of all Charpy data ≥ 95% shear is reported as the USE. In some instances, there may be data deemed 'out of family,' which are removed from the determination of the USE based on engineering judgment. However, the use of engineering judgment to remove 'out of family' data was not necessary for this material. Per Table C.6-1, no data with known shear ≥ 95% exists. Thus, the USE is set to greater than the impact energy with highest shear. If the highest shear includes multiple CVN impact energies, then the USE is set to greater than the largest impact energy. The USE for the material will be defined as the USE value with highest shear between the single and tandem wire.

Single Wire, Weld Heat # 3P4966, Linde 124 Flux, Lot # 1214 Initial USE > 88 ft-lb @ 85% Shear

Tandem Wire, Weld Heat # 3P4966, Linde 124 Flux, Lot # 1214 Initial USE > 69 ft-lb @ 65% Shear

Weld Heat # 3P4966, Linde 124 Flux, Lot # 1214 Initial USE > 88 ft-lb @ 85% Shear

C.6.3 Chemistry

The Cu and Ni wt. % chemical compositions of the V.C. Summer Unit 1 reactor vessel materials were defined by a review of the available original test documentation. If data for single or tandem wire includes multiple measurements, then the average of all available data will be used. The limiting value between the single and tandem weld will be used. The chemical compositions are summarized in Table C.6-3.

Table C.6-3 Chemistry Data for Weld Heat # 3P4966, Linde 124 Flux, Lot # 1214

Copper (wt.-%)	Nickel (wt.-%)	Source
0.03	0.90	Single Wire Check Analysis
0.03	0.88	Tandem Wire Check Analysis

Therefore, to determine the generic chemical content of the welds in the V.C. Summer reactor vessel, the below values will be used:

Weld Heat # 3P4966, Linde 124 Flux, Lot # 1214 Cu Content = 0.03 wt-%

Weld Heat # 3P4966, Linde 124 Flux, Lot # 1214 Ni Content = 0.90 wt-%

C.7 V.C. Summer Unit 1, Heat # 4P4784, Linde 124 Flux, Lot # 3930

Tables C.7-1 and C.7-2 summarize all available Charpy V-notch test data and drop-weight test data taken from the V.C. Summer Unit 1 CMTR and reactor vessel fabrication files. This data is combined with the additional data available from V.C. Summer Unit 1 surveillance program documented in WCAP-9134.

Table C.7-1 Charpy V-Notch Test Data for the Weld Heat # 4P4784, Linde 124 Flux, Lot # 3930

Temp. (°F)	CVN Impact Energy (ft-lb)	Lateral Expansion (mils)	Shear (%)
-120	12	12	7
-120	7[(a)]	7[(a)]	7
-120	9	10	8
-100	8	11	10
-100	7[(a)]	10	10
-100	8	10	10
-100	14.5	9[(a)]	18
-100	15	11	20
-50	32[(a)]	21[(a)]	34
-50	34	25	25
-25	30[(a)]	34[(a)]	21
-25	51	42	47
-20	33	36	35
-20	22[(a)]	25[(a)]	25
-20	38	39	40
10	53	51	50
10	46[(a)]	46[(a)]	60
10	50	49	50
10	73.5	74	90
10	78	68	80
10	74	63	75
10	71	61	75
10	73	65	75
10	79	70	85
10	75	65	75
10	81	71	90
10	70	68	75

Table C.7-1 Charpy V-Notch Test Data for the Weld Heat # 4P4784, Linde 124 Flux, Lot # 3930 (cont.)

Temp. (°F)	CVN Impact Energy (ft-lb)	Lateral Expansion (mils)	Shear (%)
10	81	68	90
10	58	53	75
10	64.5	52	73
10	62	50	95
40	94	78	90
40	93	61	80
40	95	55[(a)]	75
40	97	79	90
40	97	77	90
40	93	79	90
40	64[(a)]	64	75
40	70	69	75
40	66	64	80
50	84	72	96
50	71[(a)]	54[(a)]	73
80	80[(a)(b)]	67	98
80	84	65[(a)(b)]	98
125	85.5	75	100
125	89	74[(a)]	100
210	91.5[(a)]	73[(a)]	100
210	94	79	100
210	93.5	78	100
212	79[(a)]	80[(a)]	100
212	87	82	100
212	97	82	100

Notes for Table C.7-1:

(a) Minimum value used in the CVGRAPH plots in accordance with ASME Code III Subarticle NB-2331 criteria.

(b) The value fixed as the upper shelf in CVGRAPH plots.

Table C.7-2 Drop-Weight Test Data for Weld Heat # 4P4784, Linde 124 Flux, Lot # 3930

Test Temperature (°F)	Drop-Weights(a)	T_{NDT} (°F)
-10	2–NF	-50
-30	1–NF	
-40	2–NF	
-50	1–F	

Note for Table C.7-2:
(a) NF = "No Fail," F = "Fail".

C.7.1 Determination of the Initial RT_{NDT}

Using the data summarized in Tables C.7-1 and C.7-2, the initial RT_{NDT} value can be determined in accordance with the ASME Code Section III, Subarticle NB-2331 requirements. Following the requirements of ASME Code Section III, Subarticle NB-2331, Paragraph (a)(2), the minimum Charpy V-notch test data is first checked at a temperature not greater than the drop-weight T_{NDT} (or NDT) plus 60°F to determine if the material exhibits at least 50 ft-lb absorbed energy and 35 mils LE in the "weak" direction. Charpy V-notch tests were conducted at 10°F, T_{NDT} + 60°F (-50°F + 60°F = 10°F). The minimum Charpy V-notch test data at this temperature did **NOT** exhibit a minimum of 50 ft-lb absorbed energy and 35 mils lateral expansion; therefore, the Charpy V-notch tests at T_{NDT} + 60°F would **NOT** satisfy the criteria.

To precisely determine the temperature at which 50 ft-lb and 35 mils LE were obtained on the specimens, the unirradiated Charpy V-notch data may be plotted and fit using a hyperbolic tangent curve-fitting software, CVGRAPH. Only the minimum data points at each Charpy V-notch test temperature were used as input to the curve-fitting software, in accordance with ASME Code Section III, Subarticle NB-2331, Paragraph (a)(4). When plotting, the USE is fixed to the minimum Charpy impact energy or lateral expansion used in the plot which experience ≥ 95% shear. The resulting CVGRAPH figures are contained in the following pages for Charpy V-notch absorbed energy and lateral expansion.

Using these figures, the temperature at which 50 ft-lb absorbed energy and 35 mils lateral expansion were achieved may be obtained. The absorbed energy test data is more conservative than the lateral expansion test data; therefore, it becomes the dominant data set in defining initial RT_{NDT}.

$T_{50\text{ ft-lb}}$ = 10.6°F
$T_{35\text{ mils}}$ = -14.1°F
T_{Cv} = Max [$T_{50\text{ ft-lb}}$, $T_{35\text{ mil}}$] = Max [10.6°F, -14.1°F]
T_{Cv} = 10.6°F

Following the requirements of ASME Code Section III, Subarticle NB-2331, Paragraph (a)(3), the initial RT_{NDT} is the higher of T_{NDT} (determined from the drop-weight tests) and T_{Cv} (determined above) minus 60°F.

RT_{NDT} = Max [T_{NDT}, T_{CV} - 60°F]
RT_{NDT} = Max [-50°F, 10.6°F - 60°F] = Max [-50°F, -49.4°F]

Weld Heat # 4P4784, Linde 124 Flux, Lot # 3930 Initial RT_{NDT} = -49°F

C.7.2 Determination of the Initial USE

The current 10 CFR 50, Appendix G requirements specify that USE be calculated based on ASTM E185-82. Herein, USE is calculated based on an interpretation of ASTM E185-82 that is best explained by the most recent version of the ASTM E185 manual (2016 version). Using the guidelines in ASTM E185-82 and E185-16, the average of all Charpy data ≥ 95% shear is reported as the USE. In some instances, there may be data deemed 'out of family,' which are removed from the determination of the USE based on engineering judgment. However, the use of engineering judgment to remove 'out of family' data was not necessary for this material. The USE is displayed below; this value is the average of each of the impact energy values contained in Table C.7-1 with shear ≥ 95%.

Weld Heat # 4P4784, Linde 124 Flux, Lot # 3930 Initial USE
= Average (62, 84, 80, 84, 85.5, 89, 91.5, 94, 93.5, 79, 87, 97) ft-lb
= 86 ft-lb

C.7.3 Chemistry

The Cu and Ni wt. % chemical compositions of the V.C. Summer Unit 1 reactor vessel materials were defined by a review of the available original test documentation. The material's chemical properties are defined as the average of all available data. When component specific data was not available, a generic value was defined as the mean plus one standard deviation of available data from similar materials. This method is consistent with Regulatory Guide 1.99, Revision 2, which allows the mean plus one standard deviation method to be used for conservative chemistry estimates based on generic data if component specific data is not available. The chemical compositions are summarized in Table C.7-3.

Table C.7-3 Chemistry Data for Weld Heat # 4P4784, Linde 124 Flux, Lot # 3930

Copper (wt.-%)	Nickel (wt.-%)	Source
0.06	0.87	CMTR, Single Wire Analysis
0.05	0.91	CMTR, Tandem Wire Analysis & Surveillance Program Baseline Measurement
0.04	0.95	Analysis performed on irradiated specimen CW14 from Capsule U

Therefore, the chemical content will be defined as shown below going forward:

Weld Heat # 4P4784, Linde 124 Flux, Lot # 3930 Cu Content = 0.05 wt-%

Weld Heat # 4P4784, Linde 124 Flux, Lot # 3930 Ni Content = 0.91 wt-%

Figure C.7-1 Weld Heat # 4P4784, Linde 124 Flux, Lot # 3930 Plot of Measured Transverse Direction CVN Data

CVGraph 6.02: Hyperbolic Tangent Curve Printed on 9/23/2021 1:03 PM

A = 41.10 B = 38.90 C = 67.54 T0 = -5.17 D = 0.00

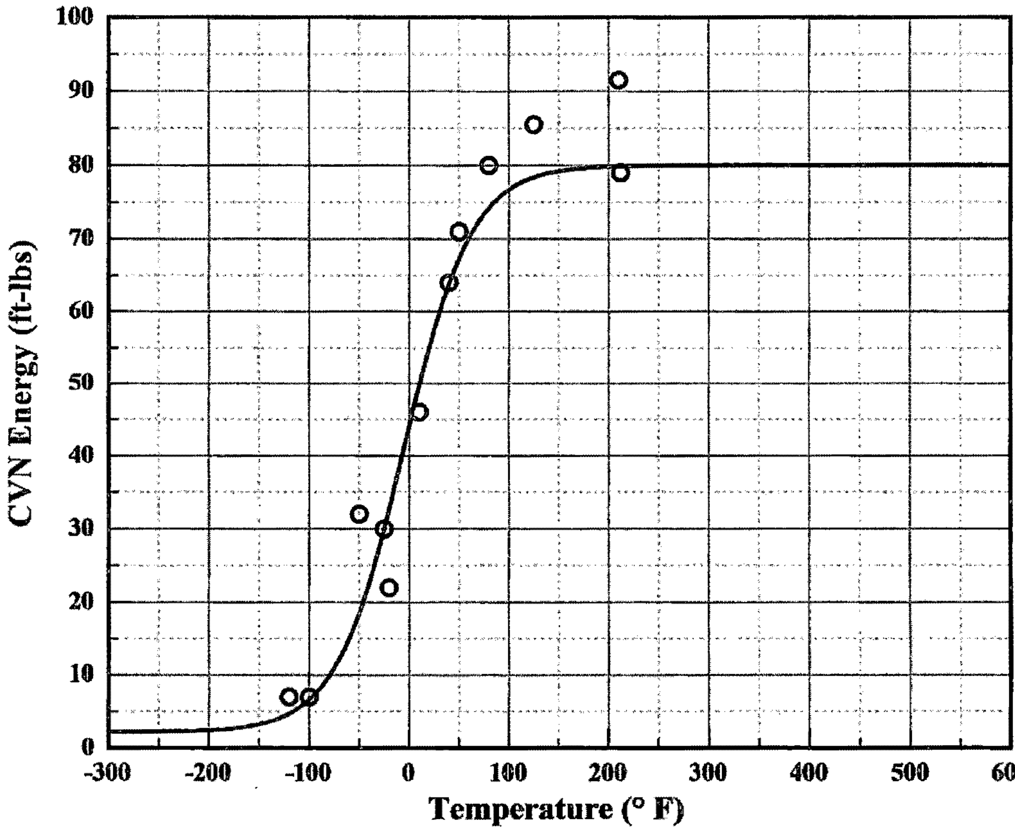
Correlation Coefficient = 0.978

Equation is A + B * [Tanh((T-T0)/(C+DT))]

Upper Shelf Energy = 80.00 (Fixed) Lower Shelf Energy = 2.20 (Fixed)

Temp@30 ft-lbs=-24.90° F Temp@35 ft-lbs=-15.80° F Temp@50 ft-lbs= 10.60° F

Plant: **V.C. Summer** Material: **SAW** Heat: **4P4784 (3930)**
Orientation: **NA** Capsule: **Unirrad** Fluence: **0.00E+000 n/cm²**



CVGraph 6.02 09/23/2021 Page 1/2

Figure C.7-1 Weld Heat # 4P4784, Linde 124 Flux, Lot # 3930 Plot of Measured Transverse Direction CVN Data (cont.)

Plant: **V.C. Summer**
Orientation: **NA**

Material: **SAW**
Capsule: **Unirrad**

Heat: **4P4784 (3930)**
Fluence: **0.00E+000 n/cm²**

Charpy V-Notch Data

Temperature (° F)	Input CVN	Computed CVN	Differential
-120	7.0	4.7	2.29
-100	7.0	6.6	0.38
-50	32.0	18.5	13.50
-25	30.0	30.0	0.01
-20	22.0	32.7	-10.69
10	46.0	49.7	-3.69
40	64.0	63.8	0.18
50	71.0	67.3	3.71
80	80.0	74.2	5.78
125	85.5	78.4	7.11
210	91.5	79.9	11.63
212	79.0	79.9	-0.87

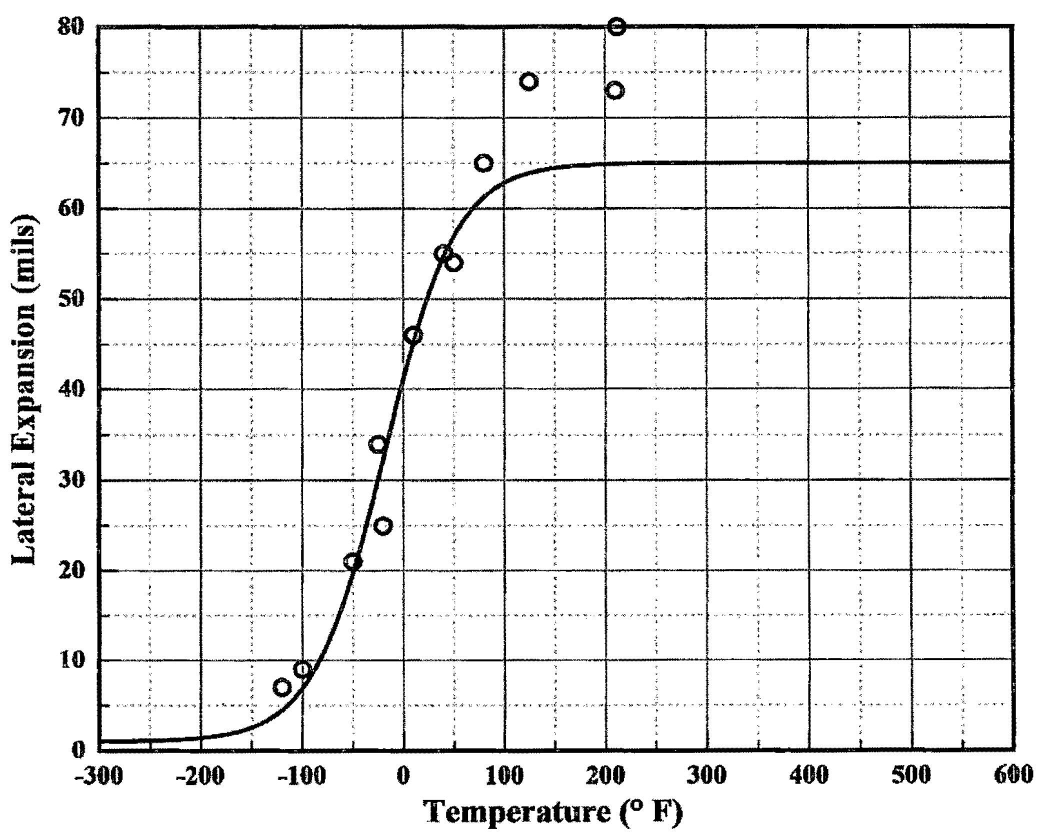
CVGraph 6.02 09/23/2021 Page 2/2

Figure C.7-2 Weld Heat # 4P4784, Linde 124 Flux, Lot # 3930
Plot of Measured Transverse Direction Lateral Expansion Data

CVGraph 6.02: Hyperbolic Tangent Curve Printed on 9/23/2021 1:42 PM
A = 33.00 B = 32.00 C = 71.15 T0 = -18.60 D = 0.00
Correlation Coefficient = 0.978
Equation is A + B * [Tanh((T-T0)/(C+DT))]

Upper Shelf L.E. = 65.00 (Fixed) Lower Shelf L.E. = 1.00 (Fixed)
Temp@35 mils=-14.10° F

Plant: **V.C. Summer** Material: **SAW** Heat: **4P4784 (3930)**
Orientation: **NA** Capsule: **Unirrad** Fluence: **0.00E+000 n/cm²**



CVGraph 6.02 09/23/2021 Page 1/2

Figure C.7-2 Weld Heat # 4P4784, Linde 124 Flux, Lot # 3930 Plot of Measured Transverse Direction Lateral Expansion Data (cont.)

Plant: **V.C. Summer**
Orientation: **NA**

Material: **SAW**
Capsule: **Unirrad**

Heat: **4P4784 (3930)**
Fluence: **0.00E+000 n/cm²**

Charpy V-Notch Data

Temperature (° F)	Input L. E.	Computed L. E.	Differential
-120	7.0	4.5	2.50
-100	9.0	6.9	2.10
-50	21.0	19.7	1.27
-25	34.0	30.1	3.87
-20	25.0	32.4	-7.37
10	46.0	45.2	0.79
40	55.0	54.7	0.34
50	54.0	56.9	-2.88
80	65.0	61.2	3.77
125	74.0	63.9	10.11
210	73.0	64.9	8.10
212	80.0	64.9	15.10

CVGraph 6.02 09/23/2021 Page 2/2

C.8 V.C. Summer Unit 1, Heat # 5P5657, Linde 124 Flux, Lot # 0931

Tables C.8-1 and C.8-2 summarize all available Charpy V-notch test data and drop-weight test data taken from the V.C. Summer Unit 1 reactor vessel fabrication files. The material properties were evaluated in two ways. 1st, the material properties were evaluated with the weld being deposited as a single wire. 2nd, the material properties were evaluated with the weld being deposited by tandem wires. For the purpose of this analysis, both will be evaluated, and the limiting property will be used.

Table C.8-1 Charpy V-Notch Test Data for the Weld Heat # 5P5657, Linde 124 Flux, Lot # 0931

Temp. (°F)	CVN Impact Energy (ft-lb)	Lateral Expansion (mils)	Shear (%)
Single Wire			
-80	39	27	5
-80	39	37	5
-80	29	32	5
-60	19	18	10
-60	20	22	10
-60	32	28	10
0	51	50	30
0	55	50	30
0	68	63	55
10	69	61	50
10	69	65	50
10	66	59	40
10	62	60	60
10	57	63	40
40	77	73	70
40	76	72	80
212	88	86	100
212	91	75	100
212	85	83	100

Table C.8-1 Charpy V-Notch Test Data for the Weld Heat # 5P5657, Linde 124 Flux, Lot # 0931 (cont.)

Temp. (°F)	CVN Impact Energy (ft-lb)	Lateral Expansion (mils)	Shear (%)
Tandem Wire			
-80	14	15	5
-80	23	22	5
-80	20	19	5
-20	42	41	20
-20	45	43	15
-20	47	44	20
-10	48	44	15
-10	46	42	20
-10	39	40	20
0	51	50	20
0	57	54	30
0	55	40	20
10	58	58	55
10	61	54	40
10	65	59	55
10	55	50	45
10	63	60	75
40	69	64	75
40	76	74	80
212	88	75	100
212	88	84	100
212	91	74	100

Table C.8-2 Drop-Weight Test Data for Weld Heat # 5P5657, Linde 124 Flux, Lot # 0931

	Test Temperature (°F)	Drop-Weights(a)	T_{NDT} (°F)
Single	-40	1–NF	-60
	-50	2–NF	
	-60	1–F	
	-70	1–F	
	-80	1–F	
Tandem	-40	1–NF	-80
	-70	2–NF	
	-80	1–F	

Note for Table C.8-2:
(a) NF = "No Fail," F = "Fail".

C.8.1 Determination of the Initial RT_{NDT}

Using the data summarized in Tables C.8-1 and C.8-2, the initial RT_{NDT} value can be determined in accordance with the ASME Code Section III, Subarticle NB-2331 requirements. Following the requirements of ASME Code Section III, Subarticle NB-2331, Paragraph (a)(2), the minimum Charpy V-notch test data is first checked at a temperature not greater than the drop-weight T_{NDT} (or NDT) plus 60°F to determine if the material exhibits at least 50 ft-lb absorbed energy and 35 mils LE.

For the single wire, the Charpy V-notch tests were conducted at 0°F, T_{NDT} + 60°F (-60°F + 60°F = 0°F). The minimum Charpy V-notch test data at this temperature exhibit a minimum of 50 ft-lb absorbed energy and 35 mils lateral expansion; therefore, the Charpy V-notch tests at T_{NDT} + 60°F satisfy the criteria. Per ASME Code Section III, Subarticle NB-2331, Paragraph (a)(2), the requirements have been met such that T_{NDT} is the initial reference temperature RT_{NDT}.

$$RT_{NDT\_single} = -60°F$$

For the tandem wire, the Charpy V-notch tests were conducted at -20°F, T_{NDT} + 60°F (-80°F + 60°F = -20°F). The minimum Charpy V-notch test data at this temperature did **NOT** exhibit a minimum of 50 ft-lb absorbed energy and 35 mils lateral expansion; therefore, the Charpy V-notch tests at T_{NDT} + 60°F would **NOT** satisfy the criteria. Since the analysis is to identify a bounding material property values for use in the extended beltline, it is not necessary precisely determine the temperature at which 50 ft-lb and 35 mils LE based on a hyperbolic tangent curve-fit. Instead, the temperature where all specimens experience greater than or equal to 50 ft-lb and 35 mils LE is used as the T50 / T35mils. From Table C.8-1, this occurred at 0°F for the tandem data. Following the requirements of ASME Code Section III, Subarticle NB-2331, Paragraph (a)(3), the initial RT_{NDT} is this temperature minus 60°F.

$RT_{NDT\_tandem} = T_{CV} - 60°F = 0°F - 60°F$
$RT_{NDT\_tandem} = -60°F$

Weld Heat # 5P5657, Linde 124 Flux, Lot # 0931 Initial RT_{NDT} = -60°F

C.8.2 Determination of the Initial USE

The current 10 CFR 50, Appendix G requirements specify that USE be calculated based on ASTM E185-82. Herein, USE is calculated based on an interpretation of ASTM E185-82 that is best explained by the most recent version of the ASTM E185 manual (2016 version). Using the guidelines in ASTM E185-82 and E185-16, the average of all Charpy data ≥ 95% shear is reported as the USE. In some instances, there may be data deemed 'out of family,' which are removed from the determination of the USE based on engineering judgment. However, the use of engineering judgment to remove 'out of family' data was not necessary for this material. As with the RT_{NDT}, both the single and tandem wire data will be evaluated separately, and the limiting value will be used.

The USE from the single and tandem wire are displayed below. These values are the average of each of the impact energy values contained in Table C.8-1 with shear ≥ 95%. The USE for the material will be defined as the minimum between the single and tandem wire USE values.

Single Wire, Weld Heat # 5P5657, Linde 124 Flux, Lot # 0931 Initial USE = Average (88, 91, 85) ft-lb
= 88 ft-lb

Tandem Wire, Weld Heat # 5P5657, Linde 124 Flux, Lot # 0931 Initial USE = Average (88, 88, 91) ft-lb
= 89 ft-lb

Weld Heat # 5P5657, Linde 124 Flux, Lot # 0931 Initial USE = Min (88, 89) ft-lb
= 88 ft-lb

C.8.3 Chemistry

The Cu and Ni wt. % chemical compositions of the V.C. Summer Unit 1 reactor vessel materials were defined by a review of the available original test documentation. If data for single or tandem wire includes multiple measurements, then the average of all available data will be used. The limiting value between the single and tandem weld will be used. The chemical compositions are summarized in Table C.8-3.

Table C.8-3 Chemistry Data for Weld Heat # 5P5657, Linde 124 Flux, Lot # 0931

Copper (wt.-%)	Nickel (wt.-%)	Source
0.07	0.71	Single Wire Check Analysis
0.04	0.89	Tandem Wire Check Analysis

Therefore, to determine the generic chemical content of the welds in the V.C. Summer reactor vessel, the below values will be used:

Weld Heat # 5P5657, Linde 124 Flux, Lot # 0931 Cu Content = 0.07 wt-%

Weld Heat # 5P5657, Linde 124 Flux, Lot # 0931 Ni Content = 0.89 wt-%

C.9 V.C. Summer Unit 1, Heat # 5P6214B, Linde 124 Flux, Lot # 0331

Tables C.9-1 and C.9-2 summarize all available Charpy V-notch test data and drop-weight test data taken from the V.C. Summer Unit 1 reactor vessel fabrication files. The material properties were evaluated in two ways. 1st, the material properties were evaluated with the weld being deposited as a single wire. 2nd, the material properties were evaluated with the weld being deposited by tandem wires. For the purpose of this analysis, both will be evaluated, and the limiting property will be used.

Table C.9-1 Charpy V-Notch Test Data for the Weld Heat # 5P6214B, Linde 124 Flux, Lot # 0331

Temp. (°F)	CVN Impact Energy (ft-lb)	Lateral Expansion (mils)	Shear (%)
Single Wire			
-70	22	17	2
-70	13	10	2
-70	11	9	2
-50	42	34	15
-50	13	11	5
-50	34	26	10
10	56	45	25
10	50	41	20
10	54	46	30
10	55	52	20
10	50	45	15
10	54	50	20
10	55	55	50
10	54	52	20
40	76	66	75
40	66	52	45
100	87	70	95
100	89	64	90
120	96	68	100
120	90	61	100
120	88	71	100

Table C.9-1 Charpy V-Notch Test Data for the Weld Heat # 5P6214B, Linde 124 Flux, Lot # 0331 (cont.)

Temp. (°F)	CVN Impact Energy (ft-lb)	Lateral Expansion (mils)	Shear (%)
Tandem Wire			
-80	17	14	2
-80	21	17	2
-40	25	20	5
-40	37	29	5
-40	29	28	5
10	50	46	50
10	61	50	40
10	64	52	35
10	61	55	30
10	37	42	20
10	54	50	25
10	47	47	15
10	68	63	45
20	75	53	55
20	69	47	60
20	78	55	55
40	82	65	75
40	80	62	75
120	100	60	100
120	96	81	100
120	97	57	100

Table C.9-2 Drop-Weight Test Data for Weld Heat # 5P6214B, Linde 124 Flux, Lot # 0331

	Test Temperature (°F)	Drop-Weights[(a)]	T_{NDT} (°F)
Single	-10	1–NF	-50
	-40	2–NF	
	-50	1–F	
Tandem	-30	2–NF	-40
	-40	1–F	
	-50	1–F	
	-60	1–F	

Note for Table C.9-2:
(a) NF = "No Fail," F = "Fail"

C.9.1 Determination of the Initial RT_{NDT}

Using the data summarized in Tables C.9-1 and C.9-2, the initial RT_{NDT} value can be determined in accordance with the ASME Code Section III, Subarticle NB-2331 requirements. Following the requirements of ASME Code Section III, Subarticle NB-2331, Paragraph (a)(2), the minimum Charpy V-notch test data is first checked at a temperature not greater than the drop-weight T_{NDT} (or NDT) plus 60°F to determine if the material exhibits at least 50 ft-lb absorbed energy and 35 mils LE.

For the single wire, the Charpy V-notch tests were conducted at 10°F, T_{NDT} + 60°F (-50°F + 60°F = 10°F). The minimum Charpy V-notch test data at this temperature exhibit a minimum of 50 ft-lb absorbed energy and 35 mils lateral expansion; therefore, the Charpy V-notch tests at T_{NDT} + 60°F satisfy the criteria. Per ASME Code Section III, Subarticle NB-2331, Paragraph (a)(2), the requirements have been met such that T_{NDT} is the initial reference temperature RT_{NDT}.

RT_{NDT_single} = -50°F

For the tandem wire, the Charpy V-notch tests were conducted at 20°F, T_{NDT} + 60°F (-40°F + 60°F = 20°F). The minimum Charpy V-notch test data at this temperature exhibit a minimum of 50 ft-lb absorbed energy and 35 mils lateral expansion; therefore, the Charpy V-notch tests at T_{NDT} + 60°F satisfy the criteria. Per ASME Code Section III, Subarticle NB-2331, Paragraph (a)(2), the requirements have been met such that T_{NDT} is the initial reference temperature RT_{NDT}.

RT_{NDT_tandem} = -40°F

Weld Heat # 5P6214B, Linde 124 Flux, Lot # 0331 Initial RT_{NDT} = -40°F

C.9.2 Determination of the Initial USE

The current 10 CFR 50, Appendix G requirements specify that USE be calculated based on ASTM E185-82. Herein, USE is calculated based on an interpretation of ASTM E185-82 that is best explained by the most recent version of the ASTM E185 manual (2016 version). Using the guidelines in ASTM E185-82 and E185-16, the average of all Charpy data ≥ 95% shear is reported as the USE. In some instances, there may be data deemed 'out of family,' which are removed from the determination of the USE based on engineering judgment. However, the use of engineering judgment to remove 'out of family' data was not necessary for this material. As with the RT_{NDT}, both the single and tandem wire data will be evaluated separately, and the limiting value will be used.

The USE from the single and tandem wire are displayed below. These values are the average of each of the impact energy values contained in Table C.9-1 with shear ≥ 95%. The USE for the material will be defined as the minimum between the single and tandem wire USE values.

Single Wire, Weld Heat # 5P6214B, Linde 124 Flux, Lot # 0331 Initial USE = Average (87, 96, 90, 88) ft-lb
= 90 ft-lb

Tandem Wire, Weld Heat # 5P6214B, Linde 124 Flux, Lot # 0331 Initial USE= Average (100, 96, 97) ft-lb
= 98 ft-lb

Weld Heat # 5P6214B, Linde 124 Flux, Lot # 0331 Initial USE = Min (90, 98) ft-lb
= 90 ft-lb

C.9.3 Chemistry

The Cu and Ni wt. % chemical compositions of the V.C. Summer Unit 1 reactor vessel materials were defined by a review of the available original test documentation. If data for single or tandem wire includes multiple measurements, then the average of all available data will be used. The limiting value between the single and tandem weld will be used. The chemical compositions are summarized in Table C.9-3.

Table C.9-3 Chemistry Data for Weld Heat # 5P6214B, Linde 124 Flux, Lot # 0331

Copper (wt.-%)	Nickel (wt.-%)	Source
0.02	0.82	Single Wire Check Analysis
0.02	0.82	
Average = 0.02	Average = 0.82	
0.014	0.70	Tandem Wire Check Analysis
0.02	0.85	
Average = 0.017	Average = 0.775	

Therefore, to determine the generic chemical content of the welds in the V.C. Summer reactor vessel, the below values will be used:

Weld Heat # 5P6214B, Linde 124 Flux, Lot # 0331 Cu Content = 0.02 wt-%

Weld Heat # 5P6214B, Linde 124 Flux, Lot # 0331 Ni Content = 0.82 wt-%

C.10 V.C. Summer Unit 1, Heat # 5P6711, Linde 124 Flux, Lot # 0342

Heat # 5P6711, Linde 124 Flux, Lot # 0342 was used in the fabrication of Shearon Harris Unit 1. Therefore, the data shown in Tables C.10-1 and C.10-2 summarize all available Charpy V-notch test data and drop-weight test data taken from the Shearon Harris Unit 1 CMTR and reactor vessel fabrication files. This data is combined with the additional data available from Shearon Harris Unit 1 surveillance program documented in WCAP-10502.

Table C.10-1 Charpy V-Notch Test Data for the Weld Heat # 5P6711, Linde 124 Flux, Lot # 0342

Temp. (°F)	CVN Impact Energy (ft-lb)	Lateral Expansion (mils)	Shear (%)
-110	7	1	15
-110	8	2	15
-100	6	7	5
-100	7	8	6
-100	7	7	4
-60	28.5	16	35
-60	34	18	40
-40	19	24	25
-40	22	29	30
-40	17	21	30
-30	28	20	55
-30	31	26	45
-30	33	26	45
0	42	36	50
0	46	41	55
0	52	45	55
10	40	41	50
10	39	46	50
10	37	38	60
10	61	52	45
10	49	41	30
10	54	47	35
10	56	48	30
10	61	53	55
10[(a)]	57[(a)]	46[(a)]	35[(a)]
10[(a)]	51[(a)]	51[(a)]	40[(a)]
10[(a)]	55	47[(a)]	40[(a)]

Table C.10-1 Charpy V-Notch Test Data for the Weld Heat # 5P6711, Linde 124 Flux, Lot # 0342 (cont.)

Temp. (°F)	CVN Impact Energy (ft-lb)	Lateral Expansion (mils)	Shear (%)
10[(a)]	61[(a)]	41[(a)]	30[(a)]
10[(a)]	42[(a)]	35[(a)]	25[(a)]
40	64	61	75
40	52	52	65
40	61	57	70
40	65	58	80
40	75	60	85
75	78	64	90
75	89	73	97
75	90	72	95
130	75	73	99
130	78	82	99
130	74	74	99
160	91	78	100
160	92	84	100
212	79	73	100
212	80	77	100
212	80	72	100
250	92	79	100
250	96	75	100
350	97	81	100

Note for Table C.10-1:

(a) Data from tandem wire specimens.

Table C.10-2 Drop-Weight Test Data for Weld Heat # 5P6711, Linde 124 Flux, Lot # 0342

Source	Test Temperature (°F)	Drop-Weights[a]	T_{NDT} (°F)
CMTR	-50	1–NF	-80
	-60	1–NF	
	-70	2–NF	
	-80	1–F	
Surv. Prog (Single)	-	-	-30
Surv. Prog (Tandem)	-	-	-20

Note for Table C.10-2:
(a) NF = "No Fail," F = "Fail"

C.10.1 Determination of the Initial RT_{NDT}

Using the data summarized in Tables C.10-1 and C.10-2, the initial RT_{NDT} value can be determined in accordance with the ASME Code Section III, Subarticle NB-2331 requirements. Following the requirements of ASME Code Section III, Subarticle NB-2331, Paragraph (a)(2), the minimum Charpy V-notch test data is first checked at a temperature not greater than the drop-weight T_{NDT} (or NDT) plus 60°F to determine if the material exhibits at least 50 ft-lb absorbed energy and 35 mils LE. The maximum T_{NDT} is used. The Charpy V-notch tests were conducted at 40°F, T_{NDT} + 60°F (-20°F + 60°F = 40°F). The minimum Charpy V-notch test data at this temperature exhibit a minimum of 50 ft-lb absorbed energy and 35 mils lateral expansion; therefore, the Charpy V-notch tests at T_{NDT} + 60°F satisfy the criteria. Per ASME Code Section III, Subarticle NB-2331, Paragraph (a)(2), the requirements have been met such that T_{NDT} is the initial reference temperature RT_{NDT}.

Weld Heat # 5P6711, Linde 124 Flux, Lot # 0342 Initial RT_{NDT} = -20°F

C.10.2 Determination of the Initial USE

The current 10 CFR 50, Appendix G requirements specify that USE be calculated based on ASTM E185-82. Herein, USE is calculated based on an interpretation of ASTM E185-82 that is best explained by the most recent version of the ASTM E185 manual (2016 version). Using the guidelines in ASTM E185-82 and E185-16, the average of all Charpy data ≥ 95% shear is reported as the USE. In some instances, there may be data deemed 'out of family,' which are removed from the determination of the USE based on engineering judgment. However, the use of engineering judgment to remove 'out of family' data was not necessary for this material. The USE is displayed below; this value is the average of each of the impact energy values contained in Table C.10-1 with shear ≥ 95% with the exception that all data at > 225°F are excluded from the calculation of USE. Per ASTM E185-16, all data specimens tested at temperatures greater than 150°F above the Charpy upper-shelf onset, i.e., ≥ 95% shear, which occurred at 75°F, shall not be included.

Weld Heat # 5P6711, Linde 124 Flux, Lot # 0342 Initial USE

= Average (89, 90, 75, 78, 74, 91, 92, 79, 80, 80) ft-lb

= 83 ft-lb

C.10.3 Chemistry

The Cu and Ni wt. % chemical compositions of the reactor vessel materials were defined by a review of the available original test documentation. The average of all available data will be used. The chemical compositions are summarized in Table C.10-3.

Table C.10-3 Chemistry Data for Weld Heat # 5P6711, Linde 124 Flux, Lot # 0342[(a)]

Copper (wt.-%)	Nickel (wt.-%)	Source
0.03	0.88	Single Wire Analysis
0.04	0.95	Tandem Wire Analysis
0.023	0.87	WCAP-10502 [HNP-1 Surveillance Report – Unirradiated]
0.026	0.94	BAW-2083 [HNP-1 Surveillance Report – Irradiated]
0.019	1.07	BAW-2083 [HNP-1 Surveillance Report – Irradiated]
0.029	0.95	BAW-2083 [HNP-1 Surveillance Report – Irradiated]

Note for Table C.10-3:

(a) Information extracted from ANP-3798NP, "Analysis of Capsule Z Duke Energy Shearon Harris Nuclear Power Plant," dated September 2019 (ADAMS Access No. ML19296C841).

Therefore, the chemical content will be defined as shown below going forward:

Weld Heat # 5P6711, Linde 124 Flux, Lot # 0342 Cu Content = 0.03 wt-%

Weld Heat # 5P6711, Linde 124 Flux, Lot # 0342 Ni Content = 0.94 wt-%

C.11 V.C. Summer Unit 1 Shielded Metal Arc Welds

Most of the SMAW Charpy tests were performed only at 2 or 3 temperature points. Table C.11-1 summarizes all available copper and nickel chemistry, Charpy V-notch test data, and drop-weight test data taken from the V.C. Summer Unit 1 reactor vessel fabrication files for the heats with Charpy tests at only a few temperature points. Those heats with Charpy data over a full temperature range are evaluated in subsequent sections.

C.11.1 Determination of the Initial RT_{NDT}

The RT_{NDT} values are determined one of two ways. First, the initial RT_{NDT} value can be determined in accordance with the ASME Code Section III, Subarticle NB-2331 requirements. Following the requirements of ASME Code Section III, Subarticle NB-2331, Paragraph (a)(2), the minimum Charpy V-notch test data is first checked at a temperature not greater than the drop-weight T_{NDT} (or NDT) plus 60°F to determine if the material exhibits at least 50 ft-lb absorbed energy and 35 mils LE. If the requirements have been met, then T_{NDT} is the initial reference temperature RT_{NDT}.

Second, if the Charpy V-notch tests at this temperature did **NOT** exhibit a minimum of 50 ft-lb absorbed energy and 35 mils lateral expansion, or the drop-weight data does not exist; NUREG-0800 BTP 5-3 Position 1.1(4) may be used as an estimate of RT_{NDT} when tests were performed at a single temperature. If all impact energies at that temperature are greater than 45 ft-lb; then the initial RT_{NDT} is equal to that temperature. If the impact energies are less than 45 ft-lb, then the initial RT_{NDT} is equal to that temperature + 20°F.

C.11.2 Determination of the Initial USE

If no data with known shear ≥ 95% exists, the USE is set to greater than the impact energy with highest shear. If the highest shear includes multiple CVN impact energies, then the USE is set to greater than the largest impact energy. The USE for the material will be defined as the USE value with highest shear between the single and tandem wire.

C.11.3 Chemistry

The Cu and Ni wt. % chemical compositions of the V.C. Summer Unit 1 reactor vessel materials were defined by a review of the available original test documentation. If data includes multiple measurements, then the average of all available data will be used. The limiting value between the single and tandem weld will be used.

Table C.11-1 SMAWs Material Properties

<table>
<tr><th>Heat Number</th><th>Lot Number</th><th>Temp (°F)</th><th>Impact Energy (ft-lb)</th><th>LE (mils)</th><th>Shear Fracture (%)</th><th>Measured Cu</th><th>Measured Ni</th><th>T_{NDT} (°F)</th><th>Initial RT_{NDT} (°F)</th><th>USE (ft-lb)</th><th>Best Est. Cu (wt. - %)</th><th>Best Est. Ni (wt. - %)</th></tr>
<tr><td rowspan="8">624263</td><td rowspan="8">E204A27A</td><td>-20</td><td>26</td><td>27</td><td>30</td><td rowspan="8">0.06</td><td rowspan="8">0.89</td><td rowspan="8">-20</td><td rowspan="8">-20</td><td rowspan="8">> 73.5 @ 75%</td><td rowspan="8">0.06</td><td rowspan="8">0.89</td></tr>
<tr><td>-20</td><td>38</td><td>34</td><td>40</td></tr>
<tr><td>-20</td><td>42</td><td>37</td><td>40</td></tr>
<tr><td>-20</td><td>50</td><td>45</td><td>50</td></tr>
<tr><td>-20</td><td>75</td><td>62</td><td>60</td></tr>
<tr><td>40</td><td>62</td><td>55</td><td>60</td></tr>
<tr><td>40</td><td>58.5</td><td>51</td><td>60</td></tr>
<tr><td>40</td><td>73.5</td><td>64</td><td>75</td></tr>
<tr><td rowspan="11">421A6811</td><td rowspan="11">F022A27A</td><td>-20</td><td>38.5</td><td>34</td><td>40</td><td rowspan="5">0.04</td><td rowspan="5">0.95</td><td rowspan="11">-20</td><td rowspan="11">-20</td><td rowspan="11">> 91 @ 75%</td><td rowspan="11">0.07</td><td rowspan="11">0.88</td></tr>
<tr><td>-20</td><td>49.5</td><td>42</td><td>45</td></tr>
<tr><td>-20</td><td>41.5</td><td>38</td><td>40</td></tr>
<tr><td>-20</td><td>38</td><td>32</td><td>40</td></tr>
<tr><td>-20</td><td>56.5</td><td>48</td><td>60</td></tr>
<tr><td>10</td><td>80</td><td>64</td><td>70</td><td rowspan="3">0.09</td><td rowspan="3">0.81</td></tr>
<tr><td>10</td><td>85</td><td>73</td><td>75</td></tr>
<tr><td>10</td><td>91</td><td>72</td><td>75</td></tr>
<tr><td>40</td><td>50</td><td>47</td><td>50</td><td>-</td><td>-</td></tr>
<tr><td>40</td><td>56.5</td><td>51</td><td>55</td><td>-</td><td>-</td></tr>
<tr><td>40</td><td>50</td><td>46</td><td>50</td><td>-</td><td>-</td></tr>
</table>

Table C.11-1 SMAWs Material Properties (cont.)

Heat Number	Lot Number	Temp (°F)	Impact Energy (ft-lb)	LE (mils)	Shear Fracture (%)	Measured Cu	Measured Ni	T_{NDT} (°F)	Initial RT_{NDT} (°F)	USE (ft-lb)	Best Est. Cu (wt. - %)	Best Est. Ni (wt. - %)
07L669	K004A27A	-20	49	41	40	0.03	1.02	N/A	-20	> 50 @ 60%	0.03	1.02
		-20	60	54	40							
		-20	55	46	35							
		-20	61	50	35							
		-20	54	49	35							
		10	50	44	50	0.03	1.02					
		10	50	44	60							
		10	54	46	40							
C3L46C	J020A27A	-20	33	30	30	0.02	0.87	-20	-20	> 57 @ 65%	0.02	0.87
		-20	34	31	35							
		-20	38	32	30							
		-20	31	28	30							
		-20	30	27	25							
		10	35	34	60	0.02	0.87					
		10	39	39	60							
		10	40	39	60							
		40	56	41	55	-	-					
		40	50	43	55	-	-					
		40	57	38	65	-	-					

Table C.11-1 SMAWs Material Properties (cont.)

<table>
<tr><th>Heat Number</th><th>Lot Number</th><th>Temp (°F)</th><th>Impact Energy (ft-lb)</th><th>LE (mils)</th><th>Shear Fracture (%)</th><th>Measured Cu</th><th>Measured Ni</th><th>T_{NDT} (°F)</th><th>Initial RT_{NDT} (°F)</th><th>USE (ft-lb)</th><th>Best Est. Cu (wt. - %)</th><th>Best Est. Ni (wt. - %)</th></tr>
<tr><td rowspan="11">422B7201</td><td rowspan="11">L030A27A</td><td>-20</td><td>52</td><td>46</td><td>35</td><td rowspan="5">0.04</td><td rowspan="5">0.9</td><td rowspan="11">-20</td><td rowspan="11">-20</td><td rowspan="11">> 66 @ 70%</td><td rowspan="11">0.04</td><td rowspan="11">0.90</td></tr>
<tr><td>-20</td><td>46</td><td>40</td><td>35</td></tr>
<tr><td>-20</td><td>43</td><td>39</td><td>30</td></tr>
<tr><td>-20</td><td>47</td><td>40</td><td>30</td></tr>
<tr><td>-20</td><td>48</td><td>40</td><td>30</td></tr>
<tr><td>10</td><td>65</td><td>52</td><td>60</td><td rowspan="3">0.04</td><td rowspan="3">0.9</td></tr>
<tr><td>10</td><td>66</td><td>55</td><td>70</td></tr>
<tr><td>10</td><td>72</td><td>56</td><td>60</td></tr>
<tr><td>40</td><td>80</td><td>55</td><td>40</td><td>-</td><td>-</td></tr>
<tr><td>40</td><td>81</td><td>54</td><td>50</td><td>-</td><td>-</td></tr>
<tr><td>40</td><td>78</td><td>57</td><td>55</td><td>-</td><td>-</td></tr>
<tr><td rowspan="8">09L853</td><td rowspan="8">A111A27A</td><td>-20</td><td>54</td><td>46</td><td>46</td><td rowspan="5">0.03</td><td rowspan="5">0.86</td><td rowspan="8">N/A</td><td rowspan="8">-20</td><td rowspan="8">> 78 @ 80%</td><td rowspan="8">0.03</td><td rowspan="8">0.86</td></tr>
<tr><td>-20</td><td>64</td><td>54</td><td>54</td></tr>
<tr><td>-20</td><td>54</td><td>45</td><td>45</td></tr>
<tr><td>-20</td><td>48</td><td>42</td><td>42</td></tr>
<tr><td>-20</td><td>65</td><td>53</td><td>53</td></tr>
<tr><td>10</td><td>78</td><td>60</td><td>70</td><td rowspan="3">0.03</td><td rowspan="3">0.86</td></tr>
<tr><td>10</td><td>78</td><td>62</td><td>80</td></tr>
<tr><td>10</td><td>79</td><td>62</td><td>60</td></tr>
</table>

Table C.11-1 SMAWs Material Properties (cont.)

Heat Number	Lot Number	Temp (°F)	Impact Energy (ft-lb)	LE (mils)	Shear Fracture (%)	Measured Cu	Measured Ni	T_{NDT} (°F)	Initial RT_{NDT} (°F)	USE (ft-lb)	Best Est. Cu (wt. - %)	Best Est. Ni (wt. - %)
08M365	G128A27A	-20	46	39	30	0.02	1.1	N/A	-20	> 51 @ 60%	0.02	1.10
		-20	50	43	35							
		-20	45	40	30							
		-20	50	43	35							
		-20	47	40	35							
		10	49	38	60	0.02	1.1					
		10	50	40	50							
		10	51	43	60							
421E0601	L117A27A	-20	33	29	40	0.03	0.87	-20	-20	> 102.5 @ 90%	0.03	0.87
		-20	37	30	35							
		-20	38	26	40							
		40	102.5	81	90	-	-					
		40	101.5	78	85	-	-					
		40	106	75	85	-	-					
09M814	L115A27A	-20	33	35	25	0.03	0.92	-20	-20	> 88 @ 80%	0.03	0.92
		-20	35	34	35							
		-20	36	31	25							
		40	78.5	68	75	-	-					
		40	81	64	75	-	-					
		40	88	74	80	-	-					

Table C.11-1 SMAWs Material Properties (cont.)

Heat Number	Lot Number	Temp (°F)	Impact Energy (ft-lb)	LE (mils)	Shear Fracture (%)	Measured Cu	Measured Ni	T_{NDT} (°F)	Initial RT_{NDT} (°F)	USE (ft-lb)	Best Est. Cu (wt. - %)	Best Est. Ni (wt. - %)
09M814	L114A27A	-20	38	34	50	0.01	0.82	-20	-20	> 92 @ 80%	0.01	0.82
		-20	38	32	50							
		-20	40	35	60							
		40	92	75	80	-	-					
		40	70	60	70	-	-					
		40	77	64	75	-	-					
623275	L121A27A	-20	28	27	40	0.05	0.84	-20	-20	> 80 @ 75%	0.05	0.84
		-20	32	28	50							
		-20	42	35	45							
		40	68.5	52	55	-	-					
		40	80	66	75							
		40	77.5	61	70							
627260	B322A27AE	-20	22	20	30	0.06	1.08	-20	-20	> 80 @ 25%	0.06	1.08
		-20	30	24	20							
		-20	31	26	20							
		-20	43	36	30							
		-20	55	44	40							
		40	77	57	25	-	-					
		40	75	77	25							
		40	80	60	25							

Table C.11-1 SMAWs Material Properties (cont.)

Heat Number	Lot Number	Temp (°F)	Impact Energy (ft-lb)	LE (mils)	Shear Fracture (%)	Measured Cu	Measured Ni	T_{NDT} (°F)	Initial RT_{NDT} (°F)	USE (ft-lb)	Best Est. Cu (wt. - %)	Best Est. Ni (wt. - %)
624039	D224A27A	-20	28	29	30	0.07	1.01	-20	-20	> 86 @ 63%	0.07	1.01
		-20	33	32	40							
		-20	34	33	40							
		-20	36	34	40							
		-20	42	42	40							
		40	80	56	45	-	-					
		40	86	63	63							
		40	79	55	55							
05P018	D211A27A	-20	29	26	30	0.09	0.90	-20	-20	> 85 @ 70%	0.09	0.90
		-20	30	26	30							
		-20	31	31	30							
		-20	36	33	40							
		-20	38	35	40							
		40	85	66	70	-	-					
		40	86	61	65							
		40	88	65	60							

Table C.11-1 SMAWs Material Properties (cont.)

Heat Number	Lot Number	Temp (°F)	Impact Energy (ft-lb)	LE (mils)	Shear Fracture (%)	Measured Cu	Measured Ni	T_{NDT} (°F)	Initial RT_{NDT} (°F)	USE (ft-lb)	Best Est. Cu (wt. - %)	Best Est. Ni (wt. - %)
421Z0611	L908A27A	-20	68	55	60	0.02	1.05	-20	-20	> 113.5 @ 85%	0.03	1.01
		-20	68.5	54	60							
		-20	66.5	55	60							
		-20	48.5	41	50							
		-20	80	64	65							
		10	66	53	70	0.03	0.96					
		10	79	58	70							
		10	80	62	80							
		40	85	61	75	-	-					
		40	98.5	76	80							
		40	113.5	79	85							
04P046	D217A27A	-20	34	23	20	0.06	0.90	N/A	0	> 40 @ 30%	0.06	0.90
		-20	36	28	20							
		-20	37	24	30							
		-20	39	20	30							
		-20	40	24	30							
624063	C228A27A	-20	37	33	30	0.03	1.00	-20	-20	> 70 @ 60%	0.03	1.00
		-20	40	34	40							
		-20	51	41	40							
		-20	57	47	50							
		-20	70	55	60							
		40	99	66	35	-	-					
		40	104	47	40							
		40	107	80	50							

C.12 V.C. Summer Unit 1, Heat # 05T776, Lot # L314A27AH

Tables C.12-1 and C.12-2 summarize all available Charpy V-notch test data and drop-weight test data taken from the V.C. Summer Unit 1 reactor vessel fabrication files.

Table C.12-1 Charpy V-Notch Test Data for the Weld Heat # 05T776, Lot # L314A27AH

Temp. (°F)	CVN Impact Energy (ft-lb)	Lateral Expansion (mils)	Shear (%)
-108	4	3	4
-108	5	4	4
-70	13	13	10
-70	14	15	10
-70	25	23	25
-30	59	33	30
-30	65	42	30
-20	38	37	15
-20	67	57	20
-20	63	53	15
-20	50	43	10
-20	72	60	20
10	65	56	40
10	84	71	50
40	101	79	80
40	108	72	75
130	103	90	100
130	126	96	100
130	127	94	100

Table C.12-2 Drop-Weight Test Data for Weld Heat # 05T776, Lot # L314A27AH

Test Temperature (°F)	Drop-Weights(a)	T_{NDT} (°F)
-60	2–NF	-70
-70	1–F	
-80	1–F	

Note for Table C.12-2:
(a) NF = "No Fail," F = "Fail".

C.12.1 Determination of the Initial RT_{NDT}

Using the data summarized in Tables C.12-1 and C.12-2, the initial RT_{NDT} value can be determined in accordance with the ASME Code Section III, Subarticle NB-2331 requirements. Following the requirements of ASME Code Section III, Subarticle NB-2331, Paragraph (a)(2), the minimum Charpy V-notch test data is first checked at a temperature not greater than the drop-weight T_{NDT} (or NDT) plus 60°F to determine if the material exhibits at least 50 ft-lb absorbed energy and 35 mils LE.

The Charpy V-notch tests were conducted at -20°F, which is less than T_{NDT} + 60°F (-70°F + 60°F = -10°F). The minimum Charpy V-notch test data at this temperature did **NOT** exhibit a minimum of 50 ft-lb absorbed energy and 35 mils lateral expansion; therefore, the Charpy V-notch tests at T_{NDT} + 60°F would **NOT** satisfy the criteria. Since the analysis is to identify a bounding material property values for use in the extended beltline, it is not necessary precisely determine the temperature at which 50 ft-lb and 35 mils LE based on a hyperbolic tangent curve-fit. Instead, the temperature where all specimens experience greater than or equal to 50 ft-lb and 35 mils LE is used as the T50 / T35mils. From Table C.12-1, this occurred at 10°F. Following the requirements of ASME Code Section III, Subarticle NB-2331, Paragraph (a)(3), the initial RT_{NDT} is this temperature minus 60°F.

$$RT_{NDT} = T_{CV} - 60°F = 10°F - 60°F$$
$$RT_{NDT} = -50°F$$

Weld Heat # 05T776, Lot # L314A27AH Initial RT_{NDT} = -50°F

C.12.2 Determination of the Initial USE

The current 10 CFR 50, Appendix G requirements specify that USE be calculated based on ASTM E185-82. Herein, USE is calculated based on an interpretation of ASTM E185-82 that is best explained by the most recent version of the ASTM E185 manual (2016 version). Using the guidelines in ASTM E185-82 and E185-16, the average of all Charpy data ≥ 95% shear is reported as the USE. In some instances, there may be data deemed 'out of family,' which are removed from the determination of the USE based on engineering judgment. However, the use of engineering judgment to remove 'out of family' data was not necessary for this material. The USE is displayed below; this value is the average of each of the impact energy values contained in Table C.12-1 with shear ≥ 95%.

Weld Heat # 05T776, Lot # L314A27AH Initial USE = Average (103, 126, 127) ft-lb
= 119 ft-lb

C.12.3 Chemistry

The Cu and Ni wt. % chemical compositions of the V.C. Summer Unit 1 reactor vessel materials were defined by a review of the available original test documentation. The material's chemical properties are defined as the average of all available data. The chemical compositions are summarized in Table C.12-3.

Table C.12-3 Chemistry Data for Weld Heat # 05T776, Lot # L314A27AH

Copper (wt.-%)	Nickel (wt.-%)	Source
0.06	0.92	CBI Wire Analysis
0.06	0.92	Chemetron verification Analysis

Therefore, to determine the generic chemical content of the welds in the V.C. Summer reactor vessel, the below values will be used:

Weld Heat # 05T776, Lot # L314A27AH Cu Content = 0.06 wt-%

Weld Heat # 05T776, Lot # L314A27AH Ni Content = 0.92 wt-%

C.13 V.C. Summer Unit 1, Heat # 422K8511, Lot # G313A27AD

Tables C.13-1 and C.13-2 summarize all available Charpy V-notch test data and drop-weight test data taken from the V.C. Summer Unit 1 reactor vessel fabrication files.

Table C.13-1 Charpy V-Notch Test Data for the Weld Heat # 422K8511, Lot # G313A27AD

Temp. (°F)	CVN Impact Energy (ft-lb)	Lateral Expansion (mils)	Shear (%)
-90	14(a)	15(a)	5
-90	17	16	5
-80	14(a)	15(a)	10
-80	16	16	10
-80	20	20	10
-40	26	26	30
-40	26(a)	24(a)	30
-40	40	33	30
-20	65	44	40
-20	74	48	50
-20	127	76	60
-20	62	52	30
-20	63	50	30
-20	83	60	35
-20	40(a)	32(a)	25
-20	49	35	25
25	107(a)	74(a)	80
25	108	80	70
40	125(a)(b)	84	100
40	125	89	100
40	140	82(a)	90
50	153	95	90
50	143(a)	81(a)	80
50	156	91	90
68	153	85(a)(b)	100
68	143(a)	96	100
68	165	91	100

Notes for Table C.13-1:

(a) Minimum value used in the CVGRAPH plots in accordance with ASME Code III Subarticle NB-2331 criteria.

(b) The value fixed as the upper shelf in CVGRAPH plots.

Table C.13-2 Drop-Weight Test Data for Weld Heat # 422K8511, Lot # G313A27AD

Test Temperature (°F)	Drop-Weights[(a)]	T_{NDT} (°F)
-39	1–NF	-80
-58	1-NF	
-70	2-NF	
-80	1–F	

Note for Table C.13-2:
(a) NF = "No Fail," F = "Fail".

C.13.1 Determination of the Initial RT_{NDT}

Using the data summarized in Tables C.13-1 and C.13-2, the initial RT_{NDT} value can be determined in accordance with the ASME Code Section III, Subarticle NB-2331 requirements. Following the requirements of ASME Code Section III, Subarticle NB-2331, Paragraph (a)(2), the minimum Charpy V-notch test data is first checked at a temperature not greater than the drop-weight T_{NDT} (or NDT) plus 60°F to determine if the material exhibits at least 50 ft-lb absorbed energy and 35 mils LE.

The Charpy V-notch tests were conducted at -20°F, T_{NDT} + 60°F (-80°F + 60°F = -20°F). The minimum Charpy V-notch test data at this temperature did **NOT** exhibit a minimum of 50 ft-lb absorbed energy and 35 mils lateral expansion; therefore, the Charpy V-notch tests at T_{NDT} + 60°F would **NOT** satisfy the criteria. However, since the majority of the Charpy V-notch tests at T_{NDT} + 60°F met the ASME Section III criterion, it was decided to plot and fit the unirradiated Charpy V-notch data using a hyperbolic tangent curve-fitting software, CVGRAPH. Only the minimum data points at each Charpy V-notch test temperature were used as input to the curve-fitting software, in accordance with ASME Code Section III, Subarticle NB-2331, Paragraph (a)(4). When plotting, the USE is fixed to the minimum Charpy impact energy or lateral expansion used in the plot which experience ≥ 95% shear. The resulting CVGRAPH figures are contained in the following pages for Charpy V-notch absorbed energy and lateral expansion.

Using these figures, the temperature at which 50 ft-lb absorbed energy and 35 mils lateral expansion were achieved may be obtained. The absorbed energy test data is more conservative than the lateral expansion test data; therefore, it becomes the dominant data set in defining initial RT_{NDT}.

$T_{50\ ft\text{-}lb}$ = -17.5°F
$T_{35\ mils}$ = -25.9°F
T_{Cv} = Max [$T_{50\ ft\text{-}lb}$, $T_{35\ mil}$] = Max [-17.5°F, -25.9°F]
T_{Cv} = -17.5°F

Following the requirements of ASME Code Section III, Subarticle NB-2331, Paragraph (a)(3), the initial RT_{NDT} is the higher of T_{NDT} (determined from the drop-weight tests) and T_{Cv} (determined above) minus 60°F.

RT_{NDT} = Max [T_{NDT}, T_{CV} - 60°F]
RT_{NDT} = Max [-80°F, -17.5°F - 60°F] = Max [-80°F, -77.5°F]

Weld Heat # 422K8511, Lot # G313A27AD Initial RT_{NDT} = -78°F

C.13.2 Determination of the Initial USE

The current 10 CFR 50, Appendix G requirements specify that USE be calculated based on ASTM E185-82. Herein, USE is calculated based on an interpretation of ASTM E185-82 that is best explained by the most recent version of the ASTM E185 manual (2016 version). Using the guidelines in ASTM E185-82 and E185-16, the average of all Charpy data ≥ 95% shear is reported as the USE. In some instances, there may be data deemed 'out of family,' which are removed from the determination of the USE based on engineering judgment. However, the use of engineering judgment to remove 'out of family' data was not necessary for this material. The USE is displayed below; this value is the average of each of the impact energy values contained in Table C.13-1 with shear ≥ 95%.

Weld Heat # 422K8511, Lot # G313A27AD Initial USE = Average (125, 125, 153, 143, 165) ft-lb
= 142 ft-lb

C.13.3 Chemistry

The Cu and Ni wt. % chemical compositions of the V.C. Summer Unit 1 reactor vessel materials were defined by a review of the available original test documentation. The material's chemical properties are defined as the average of all available data. The chemical compositions are summarized in Table C.13-3.

Table C.13-3 Chemistry Data for Weld Heat # 422K8511, Lot # G313A27AD

Copper (wt.-%)	Nickel (wt.-%)	Source
0.01	1.00	CBI Wire Analysis
0.01	1.00	Chemetron verification Analysis

Therefore, to determine the generic chemical content of the welds in the V.C. Summer reactor vessel, the below values will be used:

Weld Heat # 422K8511, Lot # G313A27AD Cu Content = 0.01 wt-%

Weld Heat # 422K8511, Lot # G313A27AD Ni Content = 1.00 wt-%

Figure C.13-1 Heat # 422K8511, Lot # G313A27AD
Plot of Measured Transverse Direction CVN Data

CVGraph 6.02: Hyperbolic Tangent Curve Printed on 9/28/2021 11:50 AM

A = 63.60 B = 61.40 C = 31.71 T0 = -10.41 D = 0.00

Correlation Coefficient = 0.986

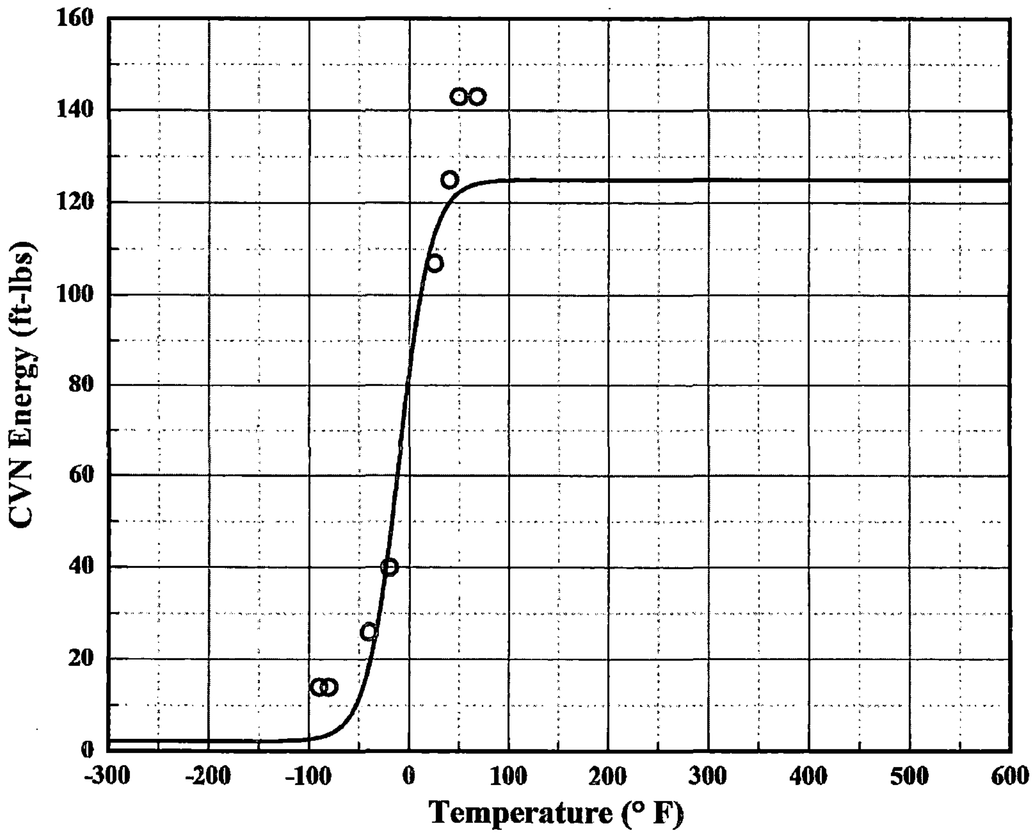
Equation is A + B * [Tanh((T-T0)/(C+DT))]

Upper Shelf Energy = 125.00 (Fixed) Lower Shelf Energy = 2.20 (Fixed)

Temp@30 ft-lbs=-29.80° F Temp@35 ft-lbs=-26.40° F Temp@50 ft-lbs=-17.50° F

Plant: **V.C. Summer** Material: **SMAW** Heat: **422K8511 (G313A27AD)**

Orientation: **NA** Capsule: **Unirrad** Fluence: **0.00E+000 n/cm²**



CVGraph 6.02 09/28/2021 Page 1/2

Figure C.13-1 Heat # 422K8511, Lot # G313A27AD Plot of Measured Transverse Direction CVN Data (cont.)

Plant: **V.C. Summer**
Orientation: **NA**

Material: **SMAW**
Capsule: **Unirrad**

Heat: **422K8511 (G313A27AD)**
Fluence: **0.00E+000 n/cm²**

Charpy V-Notch Data

Temperature (° F)	Input CVN	Computed CVN	Differential
-90	14.0	3.0	10.99
-80	14.0	3.7	10.29
-40	26.0	18.6	7.35
-20	40.0	45.6	-5.57
25	107.0	113.1	-6.11
40	125.0	120.1	4.91
50	143.0	122.3	20.66
68	143.0	124.1	18.87

CVGraph 6.02 09/28/2021 Page 2/2

Figure C.13-2 Heat # 422K8511, Lot # G313A27AD
Plot of Measured Transverse Direction Lateral Expansion Data

CVGraph 6.02: Hyperbolic Tangent Curve Printed on 9/28/2021 11:53 AM

A = 43.00 B = 42.00 C = 50.81 T0 = -16.20 D = 0.00

Correlation Coefficient = 0.987

Equation is A + B * [Tanh((T-T0)/(C+DT))]

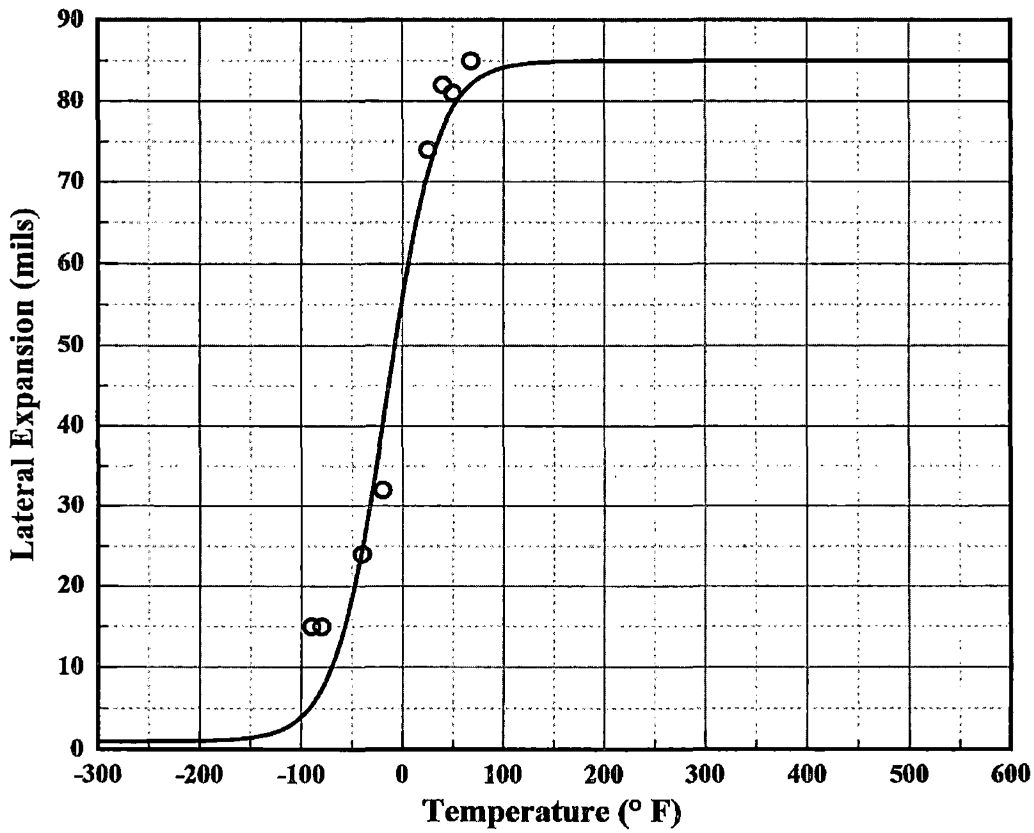
Upper Shelf L.E. = 85.00 (Fixed) Lower Shelf L.E. = 1.00 (Fixed)

Temp@35 mils=-25.90° F

Plant: **V.C. Summer**
Orientation: **NA**

Material: **SMAW**
Capsule: **Unirrad**

Heat: **422K8511 (G313A27AD)**
Fluence: **0.00E+000 n/cm²**



CVGraph 6.02 09/28/2021 Page 1/2

Figure C.13-2 Heat # 422K8511, Lot # G313A27AD
Plot of Measured Transverse Direction Lateral Expansion Data (cont.)

Plant: **V.C. Summer**
Orientation: **NA**

Material: **SMAW**
Capsule: **Unirrad**

Heat: **422K8511 (G313A27AD)**
Fluence: **0.00E+000 n/cm²**

Charpy V-Notch Data

Temperature (° F)	Input L. E.	Computed L. E.	Differential
-90	15.0	5.4	9.64
-80	15.0	7.3	7.70
-40	24.0	24.6	-0.65
-20	32.0	39.9	-7.86
25	74.0	71.1	2.86
40	82.0	76.7	5.29
50	81.0	79.2	1.78
68	85.0	82.1	2.95

CVGraph 6.02 09/28/2021 Page 2/2

C.14 V.C. Summer Unit 1, Heat # 492L4871, Lot # A421B27AE

Tables C.14-1 and C.14-2 summarize all available Charpy V-notch test data and drop-weight test data taken from the V.C. Summer Unit 1 reactor vessel fabrication files.

Table C.14-1 Charpy V-Notch Test Data for the Weld Heat # 492L4871, Lot # A421B27AE

Temp. (°F)	CVN Impact Energy (ft-lb)	Lateral Expansion (mils)	Shear (%)
-108	10	5	4
-108	11	4	4
-90	25	6	8
-90	30	6	10
-90	32	6	10
-30	19	19	20
-30	28	23	25
-30	31	25	25
-20	22	23	25
-20	26	21	25
-20	30	27	30
-20	92	63	25
-20	80	58	25
-20	66	48	20
-20	59	46	20
-20	59	42	20
-10	38	28	30
-10	41	32	30
-10	43	30	30
0	50	36	30
0	51	38	40
0	57	40	45
40	135	84	90
40	137	80	80
130	151	80	100
130	160	82	100
130	161	81	100

Table C.14-2 Drop-Weight Test Data for Weld Heat # 492L4871, Lot # A421B27AE

Test Temperature (°F)	Drop-Weights[a]	T_{NDT} (°F)
-70	2–NF	-90
-80	2–NF	
-90	1–F	

Note for Table C.14-2:
(a) NF = "No Fail," F = "Fail".

C.14.1 Determination of the Initial RT_{NDT}

Using the data summarized in Tables C.14-1 and C.14-2, the initial RT_{NDT} value can be determined in accordance with the ASME Code Section III, Subarticle NB-2331 requirements. Following the requirements of ASME Code Section III, Subarticle NB-2331, Paragraph (a)(2), the minimum Charpy V-notch test data is first checked at a temperature not greater than the drop-weight T_{NDT} (or NDT) plus 60°F to determine if the material exhibits at least 50 ft-lb absorbed energy and 35 mils LE.

The Charpy V-notch tests were conducted at -30°F, T_{NDT} + 60°F (-90°F + 60°F = -30°F). The minimum Charpy V-notch test data at this temperature did **NOT** exhibit a minimum of 50 ft-lb absorbed energy and 35 mils lateral expansion; therefore, the Charpy V-notch tests at T_{NDT} + 60°F would **NOT** satisfy the criteria. Since the analysis is to identify a bounding material property values for use in the extended beltline, it is not necessary precisely determine the temperature at which 50 ft-lb and 35 mils LE based on a hyperbolic tangent curve-fit. Instead, the temperature where all specimens experience greater than or equal to 50 ft-lb and 35 mils LE is used as the T50 / T35mils. From Table C.14-1, this occurred at 0°F. Following the requirements of ASME Code Section III, Subarticle NB-2331, Paragraph (a)(3), the initial RT_{NDT} is this temperature minus 60°F.

$$RT_{NDT} = T_{CV} - 60°F = 0°F - 60°F$$
$$RT_{NDT} = -60°F$$

Weld Heat # 492L4871, Lot # A421B27AE Initial RT_{NDT} = -60°F

C.14.2 Determination of the Initial USE

The current 10 CFR 50, Appendix G requirements specify that USE be calculated based on ASTM E185-82. Herein, USE is calculated based on an interpretation of ASTM E185-82 that is best explained by the most recent version of the ASTM E185 manual (2016 version). Using the guidelines in ASTM E185-82 and E185-16, the average of all Charpy data ≥ 95% shear is reported as the USE. In some instances, there may be data deemed 'out of family,' which are removed from the determination of the USE based on engineering judgment. However, the use of engineering judgment to remove 'out of family' data was not necessary for this material. The USE is displayed below; this value is the average of each of the impact energy values contained in Table C.14-1 with shear ≥ 95%.

Weld Heat # 492L4871, Lot # A421B27AE Initial USE = Average (151, 160, 161) ft-lb
= 157 ft-lb

C.14.3 Chemistry

The Cu and Ni wt. % chemical compositions of the V.C. Summer Unit 1 reactor vessel materials were defined by a review of the available original test documentation. The material's chemical properties are defined as the average of all available data. The chemical compositions are summarized in Table C.14-3.

Table C.14-3 Chemistry Data for Weld Heat # 492L4871, Lot # A421B27AE

Copper (wt.-%)	Nickel (wt.-%)	Source
0.04	0.95	CBI Wire Analysis
0.04	0.95	CBI verification Analysis

Therefore, to determine the generic chemical content of the welds in the V.C. Summer reactor vessel, the below values will be used:

Weld Heat # 492L4871, Lot # A421B27AE Cu Content = 0.04 wt-%

Weld Heat # 492L4871, Lot # A421B27AE Ni Content = 0.95 wt-%

C.15 V.C. Summer Unit 1, Heat # 492L4871, Lot # A421B27AF

Tables C.15-1 and C.15-2 summarize all available Charpy V-notch test data and drop-weight test data taken from the V.C. Summer Unit 1 reactor vessel fabrication files.

Table C.15-1 Charpy V-Notch Test Data for the Weld Heat # 492L4871, Lot # A421B27AF

Temp. (°F)	CVN Impact Energy (ft-lb)	Lateral Expansion (mils)	Shear (%)
-108	7	6	4
-108	9	7	3
-80	40	9	10
-80	46	10	10
-80	49	13	10
-30	50	20	20
-30	51	17	20
-30	54	21	20
-20	49	18	20
-20	51	21	20
-20	52	20	20
-20	78	55	50
-20	82	60	45
-20	105	72	40
-20	93	64	25
-20	81	60	20
-10	43	27	30
-10	55	32	35
-10	56	33	35
0	33	31	30
0	50	36	40
0	52	34	28
10	56	38	35
10	58	37	35
10	61	42	35
40	62	49	35
40	63	50	40
40	68	56	30
130	126	93	100
130	129	94	100
130	136	91	100

Table C.15-2 Drop-Weight Test Data for Weld Heat # 492L4871, Lot # A421B27AF

Test Temperature (°F)	Drop-Weights[(a)]	T_{NDT} (°F)
-70	2–NF	-80
-80	1–F	

Note for Table C.15-2:
(a) NF = "No Fail," F = "Fail".

C.15.1 Determination of the Initial RT_{NDT}

Using the data summarized in Tables C.15-1 and C.15-2, the initial RT_{NDT} value can be determined in accordance with the ASME Code Section III, Subarticle NB-2331 requirements. Following the requirements of ASME Code Section III, Subarticle NB-2331, Paragraph (a)(2), the minimum Charpy V-notch test data is first checked at a temperature not greater than the drop-weight T_{NDT} (or NDT) plus 60°F to determine if the material exhibits at least 50 ft-lb absorbed energy and 35 mils LE.

The Charpy V-notch tests were conducted at -20°F, T_{NDT} + 60°F (-80°F + 60°F = -20°F). The minimum Charpy V-notch test data at this temperature did **NOT** exhibit a minimum of 50 ft-lb absorbed energy and 35 mils lateral expansion; therefore, the Charpy V-notch tests at T_{NDT} + 60°F would **NOT** satisfy the criteria. Since the analysis is to identify a bounding material property values for use in the extended beltline, it is not necessary precisely determine the temperature at which 50 ft-lb and 35 mils LE based on a hyperbolic tangent curve-fit. Instead, the temperature where all specimens experience greater than or equal to 50 ft-lb and 35 mils LE is used as the T50 / T35mils. From Table C.15-1, this occurred at 0°F. Following the requirements of ASME Code Section III, Subarticle NB-2331, Paragraph (a)(3), the initial RT_{NDT} is this temperature minus 60°F.

$RT_{NDT} = T_{CV}$ - 60°F = 0°F - 60°F
RT_{NDT} = -60°F

Weld Heat # 492L4871, Lot # A421B27AF Initial RT_{NDT} = -60°F

C.15.2 Determination of the Initial USE

The current 10 CFR 50, Appendix G requirements specify that USE be calculated based on ASTM E185-82. Herein, USE is calculated based on an interpretation of ASTM E185-82 that is best explained by the most recent version of the ASTM E185 manual (2016 version). Using the guidelines in ASTM E185-82 and E185-16, the average of all Charpy data ≥ 95% shear is reported as the USE. In some instances, there may be data deemed 'out of family,' which are removed from the determination of the USE based on engineering judgment. However, the use of engineering judgment to remove 'out of family' data was not necessary for this material. The USE is displayed below; this value is the average of each of the impact energy values contained in Table C.15-1 with shear ≥ 95%.

Weld Heat # 492L4871, Lot # A421B27AF Initial USE = Average (126, 129, 136) ft-lb
= 130 ft-lb

C.15.3 Chemistry

The Cu and Ni wt. % chemical compositions of the V.C. Summer Unit 1 reactor vessel materials were defined by a review of the available original test documentation. The material's chemical properties are defined as the average of all available data. The chemical compositions are summarized in Table C.15-3.

Table C.15-3 Chemistry Data for Weld Heat # 492L4871, Lot # A421B27AF

Copper (wt.-%)	Nickel (wt.-%)	Source
0.03	0.98	CBI Wire Analysis
0.03	0.98	CBI verification Analysis

Therefore, to determine the generic chemical content of the welds in the V.C. Summer reactor vessel, the below values will be used:

Weld Heat # 492L4871, Lot # A421B27AF Cu Content = 0.03 wt-%

Weld Heat # 492L4871, Lot # A421B27AF Ni Content = 0.98 wt-%

C.16 V.C. Summer Unit 1, Heat # 624039, Lot # D205A27A

Tables C.16-1 and C.16-2 summarize all available Charpy V-notch test data and drop-weight test data taken from the V.C. Summer Unit 1 reactor vessel fabrication files.

Table C.16-1 Charpy V-Notch Test Data for the Weld Heat # 624039, Lot # D205A27A

Temp. (°F)	CVN Impact Energy (ft-lb)	Lateral Expansion (mils)	Shear (%)
-120	8	8	0
-120	6[(a)]	5[(a)]	0
-90	29	23	5
-90	11	9	2
-90	5[(a)]	5[(a)]	2
-60	16[(a)]	13[(a)]	5
-60	45	37	10
-60	21	19	5
-30	64	51[(a)]	20
-30	61[(a)]	52	15
-30	69	57	20
-20	41[(a)]	32[(a)]	20
-20	44	39	30
-20	49	40	30
-20	54	41	35
-20	58	45	40
40	86[(a)]	70[(a)]	50
40	96	70	80
100	116	66[(a)(b)]	95
100	106[(a)]	84	90
150	118[(a)(b)]	68[(a)]	100
150	119	98	100
150	121	91	100

Notes for Table C.16-1:

(a) Minimum value used in the CVGRAPH plots in accordance with ASME Code III Subarticle NB-2331 criteria.

(b) The value fixed as the upper shelf in CVGRAPH plots.

Table C.16-2 Drop-Weight Test Data for Weld Heat # 624039, Lot # D205A27A

Test Temperature (°F)	Drop-Weights[(a)]	T_{NDT} (°F)
-60	1–NF	-90
-80	2-NF	
-90	1-F	
-100	1–F	

Note for Table C.16-2:
(a) NF = "No Fail," F = "Fail".

C.16.1 Determination of the Initial RT_{NDT}

Using the data summarized in Tables C.16-1 and C.16-2, the initial RT_{NDT} value can be determined in accordance with the ASME Code Section III, Subarticle NB-2331 requirements. Following the requirements of ASME Code Section III, Subarticle NB-2331, Paragraph (a)(2), the minimum Charpy V-notch test data is first checked at a temperature not greater than the drop-weight T_{NDT} (or NDT) plus 60°F to determine if the material exhibits at least 50 ft-lb absorbed energy and 35 mils LE.

The Charpy V-notch tests were conducted at -30°F, T_{NDT} + 60°F (-90°F + 60°F = -30°F). The minimum Charpy V-notch test data at this temperature exhibit a minimum of 50 ft-lb absorbed energy and 35 mils lateral expansion; therefore, the Charpy V-notch tests at T_{NDT} + 60°F satisfy the criteria. However, minimum Charpy V-notch test data at a higher temperature failed to meet the ASME Section III criterion; therefore, it was decided to plot and fit the unirradiated Charpy V-notch data using a hyperbolic tangent curve-fitting software, CVGRAPH. Only the minimum data points at each Charpy V-notch test temperature were used as input to the curve-fitting software, in accordance with ASME Code Section III, Subarticle NB-2331, Paragraph (a)(4). When plotting, the USE is fixed to the minimum Charpy impact energy or lateral expansion used in the plot which experience ≥ 95% shear. The resulting CVGRAPH figures are contained in the following pages for Charpy V-notch absorbed energy and lateral expansion.

Using these figures, the temperature at which 50 ft-lb absorbed energy and 35 mils lateral expansion were achieved may be obtained. The absorbed energy test data is more conservative than the lateral expansion test data; therefore, it becomes the dominant data set in defining initial RT_{NDT}.

$T_{50\ \text{ft-lb}}$ = -16.8°F
$T_{35\ \text{mils}}$ = -33.4°F
T_{Cv} = Max [$T_{50\ \text{ft-lb}}$, $T_{35\ \text{mil}}$] = Max [-16.8°F, -33.4°F]
T_{Cv} = -16.8°F

Following the requirements of ASME Code Section III, Subarticle NB-2331, Paragraph (a)(3), the initial RT_{NDT} is the higher of T_{NDT} (determined from the drop-weight tests) and T_{Cv} (determined above) minus 60°F.

RT_{NDT} = Max [T_{NDT}, T_{CV} - 60°F]
RT_{NDT} = Max [-90°F, -16.8°F - 60°F] = Max [-90°F, -76.8°F]

Weld Heat # 624039, Lot # D205A27A Initial RT_{NDT} = -77°F

C.16.2 Determination of the Initial USE

The current 10 CFR 50, Appendix G requirements specify that USE be calculated based on ASTM E185-82. Herein, USE is calculated based on an interpretation of ASTM E185-82 that is best explained by the most recent version of the ASTM E185 manual (2016 version). Using the guidelines in ASTM E185-82 and E185-16, the average of all Charpy data ≥ 95% shear is reported as the USE. In some instances, there may be data deemed 'out of family,' which are removed from the determination of the USE based on engineering judgment. However, the use of engineering judgment to remove 'out of family' data was not necessary for this material. The USE is displayed below; this value is the average of each of the impact energy values contained in Table C.16-1 with shear ≥ 95%.

Weld Heat # 624039, Lot # D205A27A Initial USE = Average (116, 118, 119, 121) ft-lb
= 119 ft-lb

C.16.3 Chemistry

The Cu and Ni wt. % chemical compositions of the V.C. Summer Unit 1 reactor vessel materials were defined by a review of the available original test documentation. The material's chemical properties are defined as the average of all available data. The chemical compositions are summarized in Table C.16-3.

Table C.16-3 Chemistry Data for Weld Heat # 624039, Lot # D205A27A

Copper (wt.-%)	Nickel (wt.-%)	Source
0.028	0.91	CBI Wire Analysis
0.10	0.92	Action Welding Supply Co. verification Analysis

Therefore, to determine the generic chemical content of the welds in the V.C. Summer reactor vessel, the below values will be used:

Weld Heat # 624039, Lot # D205A27A Cu Content = 0.06 wt-%

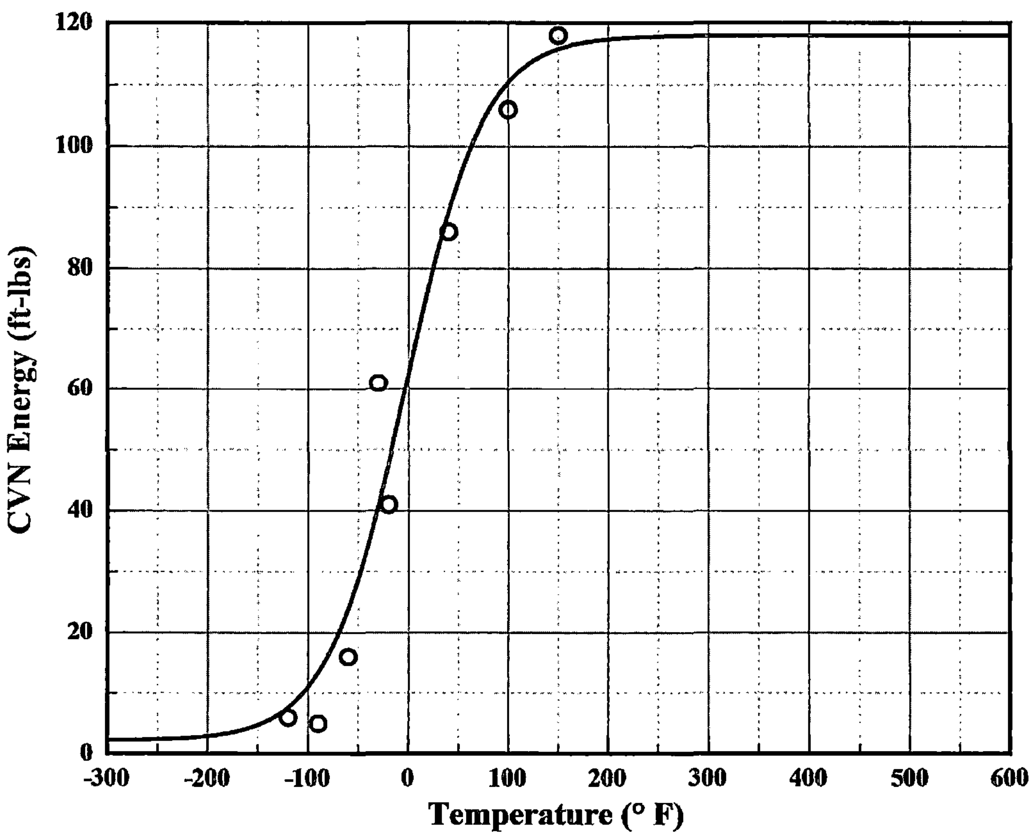
Weld Heat # 624039, Lot # D205A27A Ni Content = 0.92 wt-%

Figure C.16-1 Heat # 624039, Lot # D205A27A Plot of Measured Transverse Direction CVN Data

CVGraph 6.02: Hyperbolic Tangent Curve Printed on 9/20/2021 3:57 PM
A = 60.10 B = 57.90 C = 77.69 T0 = -3.16 D = 0.00
Correlation Coefficient = 0.978
Equation is A + B * [Tanh((T-T0)/(C+DT))]

Upper Shelf Energy = 118.00 (Fixed) Lower Shelf Energy = 2.20 (Fixed)
Temp@30 ft-lbs=-47.90° F Temp@35 ft-lbs=-39.20° F Temp@50 ft-lbs=-16.80° F

Plant: **V.C. Summer** Material: **SMAW** Heat: **624039 (D205A27A)**
Orientation: **NA** Capsule: **Unirrad** Fluence: **0.00E+000 n/cm²**



CVGraph 6.02 09/20/2021 Page 1/2

Figure C.16-1 Heat # 624039, Lot # D205A27A
Plot of Measured Transverse Direction CVN Data (cont.)

Plant: **V.C. Summer**
Orientation: **NA**

Material: **SMAW**
Capsule: **Unirrad**

Heat: **624039 (D205A27A)**
Fluence: **0.00E+000 n/cm²**

Charpy V-Notch Data

Temperature (° F)	Input CVN	Computed CVN	Differential
-120	6.0	7.7	-1.65
-90	5.0	13.4	-8.39
-60	16.0	24.0	-7.97
-30	61.0	40.9	20.14
-20	41.0	47.7	-6.75
40	86.0	89.3	-3.32
100	106.0	110.4	-4.40
150	118.0	115.8	2.20

Figure C.16-2 Heat # 624039, Lot # D205A27A
Plot of Measured Transverse Direction Lateral Expansion Data

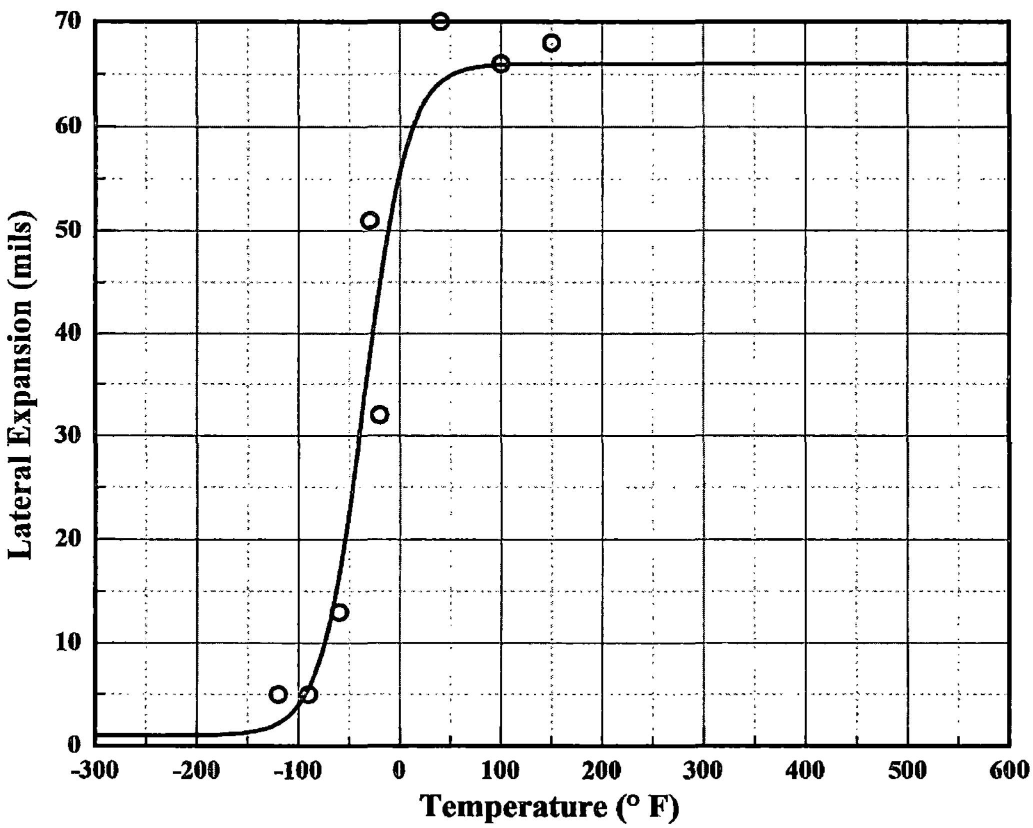
CVGraph 6.02: Hyperbolic Tangent Curve Printed on 9/20/2021 3:59 PM
A = 33.50 B = 32.50 C = 42.42 T0 = -35.42 D = 0.00
Correlation Coefficient = 0.965
Equation is A + B * [Tanh((T-T0)/(C+DT))]

Upper Shelf L.E. = 66.00 (Fixed) Lower Shelf L.E. = 1.00 (Fixed)
Temp@35 mils=-33.40° F

Plant: V.C. Summer
Orientation: NA

Material: SMAW
Capsule: Unirrad

Heat: 624039 (D205A27A)
Fluence: 0.00E+000 n/cm²



CVGraph 6.02 09/20/2021 Page 1/2

Figure C.16-2 Heat # 624039, Lot # D205A27A
Plot of Measured Transverse Direction Lateral Expansion Data (cont.)

Plant: **V.C. Summer**
Orientation: **NA**

Material: **SMAW**
Capsule: **Unirrad**

Heat: **624039 (D205A27A)**
Fluence: **0.00E+000 n/cm²**

Charpy V-Notch Data

Temperature (° F)	Input L. E.	Computed L. E.	Differential
-120	5.0	2.2	2.82
-90	5.0	5.6	-0.61
-60	13.0	16.5	-3.53
-30	51.0	37.6	13.37
-20	32.0	44.8	-12.82
40	70.0	64.2	5.81
100	66.0	65.9	0.11
150	68.0	66.0	2.01

CVGraph 6.02 09/20/2021 Page 2/2

C.17 V.C. Summer Unit 1, Heat # 626677, Lot # C301A27AF

Tables C.17-1 and C.17-2 summarize all available Charpy V-notch test data and drop-weight test data taken from the V.C. Summer Unit 1 reactor vessel fabrication files.

Table C.17-1 Charpy V-Notch Test Data for the Weld Heat # 626677, Lot # C301A27AF

Temp. (°F)	CVN Impact Energy (ft-lb)	Lateral Expansion (mils)	Shear (%)
-70	10	8	0
-70	13	12	0
-40	17	15	5
-40	20	17	5
-40	27	20	5
-20	27	24	10
-20	29	26	10
-20	24	25	30
-20	32	31	40
-20	34	32	30
-20	38	35	30
-20	44	38	30
0	43	33	20
0	22	21	15
40	53	36	25
40	51	37	25
40	54	35	25
70	66	54	75
70	70	46	75
100	83	61	90
100	89	69	95
150	90	74	100
150	92	61	100
150	102	78	100

Table C.17-2 Drop-Weight Test Data for Weld Heat # 626677, Lot # C301A27AF

Test Temperature (°F)	Drop-Weights[(a)]	T_{NDT} (°F)
-30	2–NF	-40
-40	1–F	
-60	1–F	

Note for Table C.17-2:
(a) NF = "No Fail," F = "Fail".

C.17.1 Determination of the Initial RT_{NDT}

Using the data summarized in Tables C.17-1 and C.17-2, the initial RT_{NDT} value can be determined in accordance with the ASME Code Section III, Subarticle NB-2331 requirements. Following the requirements of ASME Code Section III, Subarticle NB-2331, Paragraph (a)(2), the minimum Charpy V-notch test data is first checked at a temperature not greater than the drop-weight T_{NDT} (or NDT) plus 60°F to determine if the material exhibits at least 50 ft-lb absorbed energy and 35 mils LE.

The Charpy V-notch tests were conducted at 0°F, which is less than T_{NDT} + 60°F (-40°F + 60°F = 20°F). The minimum Charpy V-notch test data at this temperature did **NOT** exhibit a minimum of 50 ft-lb absorbed energy and 35 mils lateral expansion; therefore, the Charpy V-notch tests at T_{NDT} + 60°F would **NOT** satisfy the criteria. Since the analysis is to identify a bounding material property values for use in the extended beltline, it is not necessary precisely determine the temperature at which 50 ft-lb and 35 mils LE based on a hyperbolic tangent curve-fit. Instead, the temperature where all specimens experience greater than or equal to 50 ft-lb and 35 mils LE is used as the T50 / T35mils. From Table C.17-1, this occurred at 40°F. Following the requirements of ASME Code Section III, Subarticle NB-2331, Paragraph (a)(3), the initial RT_{NDT} is this temperature minus 60°F.

$RT_{NDT} = T_{CV}$ - 60°F = 40°F - 60°F
RT_{NDT} =-20°F

Weld Heat # 626677, Lot # C301A27AF Initial RT_{NDT} = -20°F

C.17.2 Determination of the Initial USE

The current 10 CFR 50, Appendix G requirements specify that USE be calculated based on ASTM E185-82. Herein, USE is calculated based on an interpretation of ASTM E185-82 that is best explained by the most recent version of the ASTM E185 manual (2016 version). Using the guidelines in ASTM E185-82 and E185-16, the average of all Charpy data ≥ 95% shear is reported as the USE. In some instances, there may be data deemed 'out of family,' which are removed from the determination of the USE based on engineering judgment. However, the use of engineering judgment to remove 'out of family' data was not necessary for this material. The USE is displayed below; this value is the average of each of the impact energy values contained in Table C.17-1 with shear ≥ 95%.

Weld Heat # 626677, Lot # C301A27AF Initial USE = Average (89, 90, 92, 102) ft-lb
= 93 ft-lb

C.17.3 Chemistry

The Cu and Ni wt. % chemical compositions of the V.C. Summer Unit 1 reactor vessel materials were defined by a review of the available original test documentation. The material's chemical properties are defined as the average of all available data. The chemical compositions are summarized in Table C.17-3.

Table C.17-3 Chemistry Data for Weld Heat # 626677, Lot # C301A27AF

Copper (wt.-%)	Nickel (wt.-%)	Source
0.01	0.85	CBI Wire Analysis
0.03	1.04	Chemetron verification Analysis

Therefore, to determine the generic chemical content of the welds in the V.C. Summer reactor vessel, the below values will be used:

Weld Heat # 626677, Lot # C301A27AF Cu Content = 0.02 wt-%

Weld Heat # 626677, Lot # C301A27AF Ni Content = 0.95 wt-%

C.18 V.C. Summer Unit 1, Heat # 627069, Lot # C312A27AG

Tables C.18-1 and C.18-2 summarize all available Charpy V-notch test data and drop-weight test data taken from the V.C. Summer Unit 1 reactor vessel fabrication files.

Table C.18-1 Charpy V-Notch Test Data for the Weld Heat # 627069, Lot # C312A27AG

Temp. (°F)	CVN Impact Energy (ft-lb)	Lateral Expansion (mils)	Shear (%)
-80	9	6	2
-80	8	5	2
-60	28	26	5
-60	27	24	5
-60	30	25	5
-20	31	23	30
-20	36	21	30
-20	36	26	30
-20	37	25	30
-20	38	28	40
0	72	52	35
0	64	48	35
0	78	56	45
40	68	56	40
40	86	69	75
100	117	74	95
100	107	89	90
150	114	73	100
150	117	64	100
150	112	73	100

Table C.18-2 Drop-Weight Test Data for Weld Heat # 627069, Lot # C312A27AG

Test Temperature (°F)	Drop-Weights[(a)]	T_{NDT} (°F)
-40	1–NF	-60
-50	2-NF	
-60	1–F	
-70	1–F	
-80	1–F	

Note for Table C.18-2:

(a) NF = "No Fail," F = "Fail".

C.18.1 Determination of the Initial RT_{NDT}

Using the data summarized in Tables C.18-1 and C.18-2, the initial RT_{NDT} value can be determined in accordance with the ASME Code Section III, Subarticle NB-2331 requirements. Following the requirements of ASME Code Section III, Subarticle NB-2331, Paragraph (a)(2), the minimum Charpy V-notch test data is first checked at a temperature not greater than the drop-weight T_{NDT} (or NDT) plus 60°F to determine if the material exhibits at least 50 ft-lb absorbed energy and 35 mils LE.

The Charpy V-notch tests were conducted at 0°F, T_{NDT} + 60°F (-60°F + 60°F = 0°F). The minimum Charpy V-notch test data at this temperature exhibit a minimum of 50 ft-lb absorbed energy and 35 mils lateral expansion; therefore, the Charpy V-notch tests at T_{NDT} + 60°F satisfy the criteria. Per ASME Code Section III, Subarticle NB-2331, Paragraph (a)(2), the requirements have been met such that T_{NDT} is the initial reference temperature RT_{NDT}.

Weld Heat # 627069, Lot # C312A27AG Initial RT_{NDT} = -60°F

C.18.2 Determination of the Initial USE

The current 10 CFR 50, Appendix G requirements specify that USE be calculated based on ASTM E185-82. Herein, USE is calculated based on an interpretation of ASTM E185-82 that is best explained by the most recent version of the ASTM E185 manual (2016 version). Using the guidelines in ASTM E185-82 and E185-16, the average of all Charpy data ≥ 95% shear is reported as the USE. In some instances, there may be data deemed 'out of family,' which are removed from the determination of the USE based on engineering judgment. However, the use of engineering judgment to remove 'out of family' data was not necessary for this material. The USE is displayed below; this value is the average of each of the impact energy values contained in Table C.18-1 with shear ≥ 95%.

Weld Heat # 627069, Lot # C312A27AG Initial USE = Average (117, 114, 117, 112) ft-lb
= 115 ft-lb

C.18.3 Chemistry

The Cu and Ni wt. % chemical compositions of the V.C. Summer Unit 1 reactor vessel materials were defined by a review of the available original test documentation. The material's chemical properties are defined as the average of all available data. The chemical compositions are summarized in Table C.18-3.

Table C.18-3 Chemistry Data for Weld Heat # 627069, Lot # C312A27AG

Copper (wt.-%)	Nickel (wt.-%)	Source
0.01	0.94	CBI Wire Analysis
0.03	1.04	Chemetron verification Analysis

Therefore, to determine the generic chemical content of the welds in the V.C. Summer reactor vessel, the below values will be used:

Weld Heat # 627069, Lot # C312A27AG Cu Content = 0.02 wt-%

Weld Heat # 627069, Lot # C312A27AG Ni Content = 0.99 wt-%

C.19 V.C. Summer Unit 1, Heat # 627184, Lot # C314A27AH

Tables C.19-1 and C.19-2 summarize all available Charpy V-notch test data and drop-weight test data taken from the V.C. Summer Unit 1 reactor vessel fabrication files.

Table C.19-1 Charpy V-Notch Test Data for the Weld Heat # 627184, Lot # C314A27AH

Temp. (°F)	CVN Impact Energy (ft-lb)	Lateral Expansion (mils)	Shear (%)
-70	7	5	5
-70	10	8	5
-70	6	5	5
-40	26	18	5
-40	31	21	5
-20	33	28	20
-20	34	29	30
-20	38	30	20
-20	38	27	30
-20	41	33	30
-10	46	38	25
-10	36	31	15
-10	32	30	15
0	55	46	35
0	50	37	25
0	35	26	20
10	53	40	30
10	66	45	35
10	63	46	35
40	57	45	50
40	60	39	45
40	69	59	50
80	86	70	75
80	89	73	75
120	89	62	80
120	82	60	80
180	107	94	100
180	101	60	100
180	97	77	100

Table C.19-2 Drop-Weight Test Data for Weld Heat # 627184, Lot # C314A27AH

Test Temperature (°F)	Drop-Weights[(a)]	T_{NDT} (°F)
-60	2–NF	
-70	1-NF, 1–F	-70
-80	1–F	

Note for Table C.19-2:
(a) NF = "No Fail," F = "Fail".

C.19.1 Determination of the Initial RT_{NDT}

Using the data summarized in Tables C.19-1 and C.19-2, the initial RT_{NDT} value can be determined in accordance with the ASME Code Section III, Subarticle NB-2331 requirements. Following the requirements of ASME Code Section III, Subarticle NB-2331, Paragraph (a)(2), the minimum Charpy V-notch test data is first checked at a temperature not greater than the drop-weight T_{NDT} (or NDT) plus 60°F to determine if the material exhibits at least 50 ft-lb absorbed energy and 35 mils LE.

The Charpy V-notch tests were conducted at -10°F, T_{NDT} + 60°F (-70°F + 60°F = -10°F). The minimum Charpy V-notch test data at this temperature did **NOT** exhibit a minimum of 50 ft-lb absorbed energy and 35 mils lateral expansion; therefore, the Charpy V-notch tests at T_{NDT} + 60°F would **NOT** satisfy the criteria. Since the analysis is to identify a bounding material property values for use in the extended beltline, it is not necessary precisely determine the temperature at which 50 ft-lb and 35 mils LE based on a hyperbolic tangent curve-fit. Instead, the temperature where all specimens experience greater than or equal to 50 ft-lb and 35 mils LE is used as the T50 / T35mils. From Table C.19-1, this occurred at 10°F. Following the requirements of ASME Code Section III, Subarticle NB-2331, Paragraph (a)(3), the initial RT_{NDT} is this temperature minus 60°F.

$RT_{NDT} = T_{CV}$ - 60°F = 10°F - 60°F
RT_{NDT} = -50°F

Weld Heat # 627184, Lot # C314A27AH Initial RT_{NDT} = -50°F

C.19.2 Determination of the Initial USE

The current 10 CFR 50, Appendix G requirements specify that USE be calculated based on ASTM E185-82. Herein, USE is calculated based on an interpretation of ASTM E185-82 that is best explained by the most recent version of the ASTM E185 manual (2016 version). Using the guidelines in ASTM E185-82 and E185-16, the average of all Charpy data ≥ 95% shear is reported as the USE. In some instances, there may be data deemed 'out of family,' which are removed from the determination of the USE based on engineering judgment. However, the use of engineering judgment to remove 'out of family' data was not necessary for this material. The USE is displayed below; this value is the average of each of the impact energy values contained in Table C.19-1 with shear ≥ 95%.

Weld Heat # 627184, Lot # C314A27AH Initial USE **= Average (107, 101, 97) ft-lb**
= 102 ft-lb

C.19.3 Chemistry

The Cu and Ni wt. % chemical compositions of the V.C. Summer Unit 1 reactor vessel materials were defined by a review of the available original test documentation. The material's chemical properties are defined as the average of all available data. The chemical compositions are summarized in Table C.19-3.

Table C.19-3 Chemistry Data for Weld Heat # 627184, Lot # C314A27AH

Copper (wt.-%)	Nickel (wt.-%)	Source
0.03	1.01	CBI Wire Analysis
0.03	1.01	Chemetron verification Analysis

Therefore, to determine the generic chemical content of the welds in the V.C. Summer reactor vessel, the below values will be used:

Weld Heat # 627184, Lot # C314A27AH Cu Content = 0.03 wt-%

Weld Heat # 627184, Lot # C314A27AH Ni Content = 1.01 wt-%

Enclosure 4
Attachment 2

WCAP-13207-NP, REVISION 4
(Redacted version of WCAP-13206-P)

Virgil C. Summer (VCSNS) Unit 1
Dominion Energy South Carolina, Inc. (DESC)

WCAP-13207
Revision 4

April 2022

Technical Justification for Eliminating Large Primary Loop Pipe Rupture as the Structural Design Basis for the Virgil C. Summer Nuclear Power Plant

Westinghouse

WCAP-13207
Revision 4

Technical Justification for Eliminating Large Primary Loop Pipe Rupture as the Structural Design Basis for the Virgil C. Summer Nuclear Power Plant

April 2022

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TABLE OF CONTENTS

1.0 Introduction 1-1
- 1.1 Purpose 1-1
- 1.2 Scope and Objective 1-1
- 1.3 Background Information 1-2
- 1.4 References 1-3

2.0 Operation and Stability of the Reactor Coolant System 2-1
- 2.1 Stress Corrosion Cracking 2-1
- 2.2 Water Hammer 2-2
- 2.3 Low Cycle and High Cycle Fatigue 2-3
- 2.4 References 2-3

3.0 Pipe Geometry and Loading 3-1
- 3.1 Introduction to Methodology 3-1
- 3.2 Calculation of Loads and Stresses 3-1
- 3.3 Loads for Leak Rate Evaluation 3-2
- 3.4 Load Combination for Crack Stability Analyses 3-3
- 3.5 References 3-3

4.0 Material Characterization 4-1
- 4.1 Primary Loop Pipe and Fittings Materials 4-1
- 4.2 Tensile Properties 4-1
- 4.3 Fracture Toughness Properties 4-2
- 4.4 Reference 4-6

5.0 Critical Location and Evaluation Criteria 5-1
- 5.1 Critical Locations 5-1
- 5.2 Fracture Criteria 5-1

6.0 Leak Rate Predictions 6-1
- 6.1 Introduction 6-1
- 6.2 General Considerations 6-1
- 6.3 Calculation Method 6-1
- 6.4 Leak Rate Calculations 6-2
- 6.5 References 6-2

7.0 Fracture Mechanics Evaluation 7-1
- 7.1 Local Failure Mechanism 7-1
- 7.2 Global Failure Mechanism 7-1
- 7.3 Results of Crack Stability Evaluation 7-2
- 7.4 SG and RPV Nozzle Alloy 82/182 Welds 7-4
- 7.5 References 7-5

8.0 Fatigue Crack Growth Analysis 8-1
- 8.1 References 8-3

9.0 Assessment of Margins 9-1

10.0 Conclusions 10-1

APPENDIX A: Limit Moment A-1

LIST OF TABLES

Table 3-1 Dimensions, Normal Loads and Normal Stresses for VCSNS Unit 1 3-4

Table 3-2 Dimensions, Faulted Loads and Faulted Stresses for VCSNS Unit 1 3-5

Table 4-1 Measured Room Temperature Tensile Properties for VCSNS Unit 1 Primary Loop Piping...4-7

Table 4-2 Measured Room Temperature Tensile Properties for VCSNS Unit 1 Primary Loop Fittings (Elbows)...4-8

Table 4-3 Measured Room Temperature Tensile Properties for Hot Leg Loop A Inconel 152 Weld at Location 1 4-9

Table 4-4 Mechanical Properties for VCSNS Unit 1 Materials at Operating Temperatures.................4-10

Table 4-5 Chemistry and Fracture Toughness Properties of the SA351 CF8A Material Heats of VCSNS Unit 1 4-11

Table 4-6 Enveloped J_{Ic}, J_{max}, T_{mat} for the SA351 CF8A Material Heats from Revision 1 and Revision 2 of NUREG/CR-4513...4-12

Table 4-7 Fracture Toughness Properties of SA351 CF8A for VCSNS Unit 1 Primary Loops for Leak-Before-Break Evaluation at Critical Locations..4-13

Table 6-1 Flaw Sizes Yielding a Leak Rate of 10 gpm at the Critical Locations with SA376-TP304N Material..6-3

Table 6-2 Flaw Sizes Yielding a Leak Rate of 10 gpm at the Elbow Critical Locations with SA 351-CF8A Material 6-3

Table 6-3 Flaw Sizes Yielding a Leak Rate of 10 gpm at the Critical Locations with Alloy 82/182 and Inconel 152 Materials 6-4

Table 7-1 VCSNS Unit 1 Stability Results Based on Elastic-Plastic J-Integral Evaluations for SA 351-CF8A..7-6

Table 7-2 VCSNS Unit 1 Stability Results Based on Limit Load for SA376-TP304N Material 7-6

Table 7-3 VCSNS Unit 1 Stability Results Based on Limit Load for SA351-CF8A Materials.............7-7

Table 7-4 VCSNS Unit 1 Stability Results Based on Limit Load for Alloy 82/182 and Inconel 152 Materials 7-7

Table 8-1 Summary of Reactor Vessel Transients for VCSNS Unit 1 8-4

Table 8-2 Typical Fatigue Crack Growth at [.........................]^{a,c,e} (40, 60 and 80-Years) 8-5

Table 9-1 VCSNS Unit 1 Leakage Flaw Sizes, Critical Flaw Sizes, and Margins based on Limit Load Evaluation for SA376-TP304N Material...9-2

Table 9-2 VCSNS Unit 1 Leakage Flaw Sizes, Critical Flaw Sizes, and Margins based on Limit Load and J-Integral Evaluation for SA351-CF8A Material..9-2

Table 9-3 VCSNS Unit 1 Leakage Flaw Sizes, Critical Flaw Sizes, and Margins based on Limit Load Evaluation for Alloy 82/182 and Inconel 152 Materials.......................................9-3

LIST OF FIGURES

Figure 3-1 Hot Leg Coolant Pipe 3-6

Figure 3-2 Schematic Diagram of VCSNS Unit 1 Primary Loop Showing Weld Locations 3-7

Figure 4-1 Pre-Service J vs. Δa for SA351-CF8M Cast Stainless Steel at 600°F 4-14

Figure 6-1 Analytical Predictions of Critical Flow Rates of Steam-Water Mixtures 6-5

Figure 6-2 [ ][a,c,e] Pressure Ratio as a Function of L/D 6-6

Figure 6-2 [ ][a,c,e] Pressure Ratio as a Function of L/D 6-6

Figure 6-3 Idealized Pressure Drop Profile Through a Postulated Crack 6-7

Figure 7-1 [ ][a,c,e] Stress Distribution 7-8

Figure 8-1 Typical Cross-Section of [ ][a,c,e] 8-6

Figure 8-2 Reference Fatigue Crack Growth Curves for [ ][a,c,e] 8-7

Figure 8-3 Reference Fatigue Crack Growth Law for [ ][a,c,e] in a Water Environment at 600°F 8-8

Figure A-1 Pipe with a Through-Wall Crack in Bending A-2

EXECUTIVE SUMMARY

Westinghouse performed analysis for the Leak-Before-Break (LBB) of Virgil C. Summer Nuclear Station (VCSNS), Unit 1, primary loop piping in 1992. The results of the analysis were documented in WCAP-13206 (Reference 1-1) and approved by the United States Nuclear Regulatory Commission (U.S. NRC) in a letter dated January 11, 1993 (Reference 1-2). Westinghouse also performed an evaluation of the LBB of VCSNS primary loop piping in 1993 to account the effects of Steam Generator Replacement/Uprating program and the results were documented in WCAP-13605 (Reference 1-3).

Revision 1:

Westinghouse performed an LBB evaluation of the primary loop piping of VCSNS Unit 1 primary loop piping in 1997 due to the Steam Generator (SG) snubber elimination program. An evaluation was performed to account the combined effects of SG snubber elimination and revised Replacement Steam Generators (RSGs) weight and center of gravity (CG). Revision 1 was to document the results of the latest LBB evaluation and was a general revision of the original WCAP-13206. WCAP-13206 Revision 1 superseded the WCAP-13605 (Reference 1-3).

In WCAP-13206 Revision 1 it was shown that the primary loops are highly resistant to stress corrosion cracking and high and low cycle fatigue. Water hammer is mitigated by system design and operating procedures.

The primary loops were extensively examined. The as-built geometries for the pipe and elbows and loadings were obtained. The materials were evaluated using the Certified Materials Test Reports (CMTRs). Mechanical properties were determined at operating temperatures. Since the piping systems are fabricated from cast stainless steel, fracture toughness values considering thermal aging were determined for each heat of material.

Based on loading, pipe geometry and fracture toughness considerations, enveloping critical locations were determined at which Leak-Before-Break crack stability evaluations were made. Through-wall flaw sizes were postulated which would leak at a rate of ten times the leakage detection system capability of the plant. Large margins for such flaw sizes were shown against flaw instability. Finally, fatigue crack growth was shown not to be an issue for the primary loops.

The effects of RSG and SG snubber elimination on the continued applicability of LBB for the reactor coolant loop piping of VCSNS Unit 1 were evaluated.

Revision 2:

For the hot leg of loop A at the reactor vessel outlet nozzle location, a replacement spool piece and welds (stainless steel and Inconel) were installed. The purpose of Revision 2 of WCAP-13206 was to reconcile the LBB analysis for this new configuration.

It was demonstrated that the previous LBB conclusions (References 1-1 and 1-3) analyses remained valid, and the dynamic effects of reactor coolant system primary loop pipe breaks need not be considered in the structural design basis of the VCSNS Unit 1 after the RSG, SG snubber elimination and the new configuration of hot leg loop A.

Revision 3:

A mechanical stress improvement process (MSIP) was applied at the reactor vessel outlet nozzle location on loops B and C to mitigate the effects of primary water stress corrosion cracking (PWSCC) of the nozzle to safe end welds at these locations. Application of MSIP introduced axial displacements in the primary loop piping. The LBB evaluation was updated to account for the effects of MSIP and new results are presented to demonstrate that dynamic effects of reactor coolant system primary loop pipe breaks need not be considered in the structural design basis of the VCSNS Unit 1.

Additionally, the evaluation was updated again in 2004 to demonstrate that LBB remains valid for the 60-year license renewal period.

Revision 3 of WCAP-13206 is issued to document the updated results for consideration of MSIP and the license renewal period.

Revision 4:

The purpose of Revision 4 of WCAP-13206 is to document the LBB evaluations for the VCSNS Unit 1 primary reactor loop piping due to the subsequent license renewal (SLR) program for the plant operation extension into 80 years.

For the SLR program, this report demonstrates that the conclusions reached in WCAP-13206 Revision 3 remain applicable in the structural design basis for the 80-year plant life.

In addition, this report reviews the dissimilar metal (DM) weld locations at the reactor pressure vessel nozzles and the steam generator (SG) nozzles, which are susceptible to primary water stress corrosion cracking (PWSCC) effect to confirm that those locations have been appropriately mitigated and evaluated for LBB. Except for the reactor pressure vessel inlet nozzle (RPVIN) welds, all DM weld locations (reactor pressure vessel outlet nozzles (RPVONs), steam generator inlet nozzles (SGINs) and steam generator outlet nozzles (SGONs)) have been mitigated from PWSCC effect. It should be noted that the mitigative measures were taken for the DM welds at the SGIN and SGON to safe-end locations of VCSNS Unit 1 by installing Alloy 152 inlays on the inside surface of the dissimilar metal as a protective barrier for the Alloy 82 weld against PWSCC effect.

All critical locations are evaluated, including the mitigated RPVON and SG nozzle to safe-end locations, and unmitigated RPVIN safe-end locations, to reconfirm that the LBB evaluation conclusions remain valid for 80-year plant life SLR program. As shown in the performed LBB evaluation, the presence of Alloy 82/182 at the RPVIN (location 12) is also acceptable, since all the recommended LBB margins are satisfied.

The analysis accounts for the effects of the MSIP on the loops B and C reactor vessel outlet nozzles and the updated configuration replacement spool piece and Inconel 152 weld at the loop A reactor vessel outlet nozzle as previously evaluated in Revision 2 and Revision 3 of WCAP-13206. Therefore, the LBB analysis and results for Loop A are presented separately from the LBB analysis and results for Loops B and C. Note that bounding loads from Loops B and C are considered in the LBB evaluation.

The analysis also reconciles the impact on the RCL leak-before-break (LBB) loads due to the replacement reactor vessel closure head (RRVCH) and integrated head assembly (IHA) package. Since only the LOCA loads were mainly impacted with the RRVCH and IHA package, it is concluded that the impact on the LBB analysis is insignificant.

The results of the DM weld evaluation show that the presence of Alloy 82 or Alloy 82/182 is no longer a concern for primary water stress corrosion cracking at these locations.

Non-Proprietary Version:

WCAP-13207, Revision 0, was generated as the non-proprietary version of WCAP-13206, Revision 0 (Reference 1-1). Non-proprietary version of WCAP-13206, Revision 1 and Revision 2 were never requested and never generated. Non-proprietary version of WCAP-13206, Revision 3, was issued as WCAP-13207, Revision 3. This report is being generated as the non-proprietary version of WCAP-13206, Revision 4 and is being issued as WCAP-13207, Revision 4 to maintain consistent revision numbering between the proprietary and non-proprietary versions.

1.0 INTRODUCTION

1.1 PURPOSE

This report applies to the VCSNS Unit 1 Reactor Coolant System (RCS) primary loop piping only and does not apply to any branch lines connected to the primary loop piping (e.g., surge, accumulator, RHR and safety injection branch lines). It is intended to demonstrate that for the VCSNS Unit 1, RCS primary loop pipe breaks need not be considered in the structural design basis after the RSG, SG snubber elimination, replacement of the loop A hot leg spool piece, application of MSIP on the loops B and C reactor vessel outlet nozzle, inlay application of Alloy 152 at the SGINs and SGONs, and the SLR for the plant operation extension from 60-years to 80-years. The approach taken has been accepted by the U.S. NRC per Generic Letter 84-04 (Reference 1-4).

1.2 SCOPE AND OBJECTIVE

The purpose of this investigation is to demonstrate Leak-Before-Break for the VCSNS Unit 1 primary loops piping for 80-years of plant service. The recommendations and criteria proposed in SRP 3.6.3 (Reference 1-5 and 1-11) are used in this evaluation. The criteria and resulting steps of the evaluation procedure can be briefly summarized as follows:

1. Calculate the applied loads. Identify the locations at which the highest stress occurs.
2. Identify the materials and the associated material properties.
3. Postulate a through-wall flaw at the governing locations. The size of the flaw should be large enough so that the leakage is assured of detection with margin using the installed leak detection equipment when the pipe is subjected to normal operating loads. A margin of 10 is demonstrated between the calculated leak rate and the leak detection capability.
4. Using maximum faulted loads, demonstrate that there is a margin of at least 2 between the leakage flaw size and the critical flaw size.
5. Review the operating history to ascertain that operating experience has indicated no particular susceptibility to failure from the effects of corrosion, water hammer, or low and high cycle fatigue.
6. For the materials actually used in the plant, provide representative material properties including toughness and tensile test data. Evaluate long term effects such as thermal aging.
7. Demonstrate margin on the calculated applied load value; margin of 1.4 using algebraic summation of faulted loads or margin of 1.0 using absolute summation of faulted loads.
8. Perform an assessment of fatigue crack growth. Show that a through-wall crack will not result.

This report provides a fracture mechanics demonstration of primary loop integrity for the VCSNS Unit 1 consistent with the NRC position for exemption from consideration of dynamic effects.

It should be noted that the terms "flaw" and "crack" have the same meaning and are used interchangeably. "Governing location" and "critical location" are also used interchangeably throughout the report.

1.3 BACKGROUND INFORMATION

Westinghouse has performed considerable testing and analysis to demonstrate that RCS primary loop pipe breaks can be eliminated from the structural design basis of all Westinghouse plants. The concept of eliminating pipe breaks in the RCS primary loop was first presented to the NRC in 1978 in WCAP-9283 (Reference 1-6). That topical report employed a deterministic fracture mechanics evaluation and a probabilistic analysis to support the elimination of RCS primary loop pipe breaks. That approach was then used as a means of addressing Generic Issue A-2 and Asymmetric Loss of Coolant Accident (LOCA) Loads.

Westinghouse performed additional testing and analysis to justify the elimination of RCS primary loop pipe breaks. This material was provided to the NRC along with Letter Report NS-EPR-2519 (Reference 1-7).

The NRC funded research through Lawrence Livermore National Laboratory (LLNL) to address this same issue using a probabilistic approach. As part of the LLNL research effort, Westinghouse performed extensive evaluations of specific plant loads, material properties, transients, and system geometries to demonstrate that the analysis and testing previously performed by Westinghouse and the research performed by LLNL applied to all Westinghouse plants (References 1-8 and 1-9). The results from the LLNL study were released at a March 28, 1983, Advisory Committee on Reactor Safeguards (ACRS) Subcommittee meeting. These studies which are applicable to all Westinghouse plants east of the Rocky Mountains determined the mean probability of a direct LOCA (RCS primary loop pipe break) to be 4.4 x 10^{-12} per reactor year and the mean probability of an indirect LOCA to be 10^{-7} per reactor year. Thus, the results previously obtained by Westinghouse (Reference 1-6) were confirmed by an independent NRC research study.

Based on the studies by Westinghouse, LLNL, the ACRS, and the Atomic Industrial Forum (AIF), the NRC completed a safety review of the Westinghouse reports submitted to address asymmetric blowdown loads that result from a number of discrete break locations on the pressurized water reactor (PWR) primary systems. The NRC Staff evaluation (Reference 1-4) concludes that an acceptable technical basis has been provided so that asymmetric blowdown loads need not be considered for those plants that can demonstrate the applicability of the modeling and conclusions contained in the Westinghouse response or can provide an equivalent fracture mechanics demonstration of the primary coolant loop integrity. In a more formal recognition of LBB methodology applicability for PWRs, the NRC appropriately modified 10 CFR 50, General Design Criterion 4, "Requirements for Protection Against Dynamic Effects of Postulated Pipe Rupture" (Reference 1-10).

This report provides a fracture mechanics demonstration of primary loop integrity for the VCSNS Unit 1 consistent with the NRC position for exemption from consideration of dynamic effects. The re-evaluations were performed to ensure that the LBB evaluation conclusions remain valid for 80-year plant life in the SLR program.

Several computer codes are used in the evaluations. The LBB computer programs are under Configuration Control which has requirements conforming to Standard Review Plan 3.9.1. The computer codes used in this evaluation for leak rate and fracture mechanics calculations have been validated and used for all the LBB applications by Westinghouse.

1.4 REFERENCES

1-1 WCAP-13206, Revision 0, "Technical Justification for Eliminating Large Primary Loop Pipe Rupture as the Structural Design Basis for the Virgil C. Summer Nuclear Power Plant," April 1992.

1-2 NRC Docket No. 50-395 dated January 11, 1993, "Safety Evaluation of Request to use Leak-Before-Break for Reactor Coolant System Piping-Virgil C. Summer Nuclear Station Unit No. 1 (TAC No. M83971)."

1-3 WCAP-13605, Revision 0, "Primary Loop Leak-Before-Break Reconciliation to Account for the Effects of Steam Generator Replacement/Uprating," March 1993.

1-4 U.S. NRC Generic Letter 84-04, Subject "Safety Evaluation of Westinghouse Topical Reports Dealing with Elimination of Postulated Pipe Breaks in PWR Primary Main Loops," February 1, 1984.

1-5 Standard Review Plan; public comments solicited; 3.6.3 Leak-Before-Break Evaluation Procedures; Federal Register/Vol. 52, No. 167/Friday, February 28, 1987/Notices, pp. 32626-32633.

1-6 WCAP-9283, "The Integrity of Primary Piping Systems of Westinghouse Nuclear Power Plants During Postulated Seismic Events," March 1978.

1-7 Letter Report NS-EPR-2519, Westinghouse (E. P. Rahe) to NRC (D. G. Eisenhut), Westinghouse Proprietary Class 2, November 10, 1981.

1-8 Letter from Westinghouse (E. P. Rahe) to NRC (W. V. Johnston), April 25, 1983.

1-9 Letter from Westinghouse (E. P. Rahe) to NRC (W. V. Johnston), July 25, 1983.

1-10 Nuclear Regulatory Commission, 10 CFR 50, Modification of General Design Criteria 4 Requirements for Protection Against Dynamic Effects of Postulated Pipe Ruptures, Final Rule, Federal Register/Vol. 52, No. 207/Tuesday, October 27, 1987/Rules and Regulations, pp. 41288-41295.

1-11 NUREG-0800 Revision 1, March 2007, Standard Review Plan: 3.6.3 Leak-Before-Break Evaluation Procedures.

2.0 OPERATION AND STABILITY OF THE REACTOR COOLANT SYSTEM

2.1 STRESS CORROSION CRACKING

The Westinghouse reactor coolant system primary loops have an operating history that demonstrates the inherent operating stability characteristics of the design. This includes a low susceptibility to cracking failure from the effects of corrosion (e.g., intergranular stress corrosion cracking (IGSCC)). This operating history totals over 1400 reactor-years, including 16 plants each having over 30 years of operation, 10 other plants each with over 25 years of operation, 11 plants each with over 20 years of operation, and 12 plants each with over 15 years of operation.

In 1978, the United States Nuclear Regulatory Commission (U.S. NRC) formed the second Pipe Crack Study Group. (The first Pipe Crack Study Group (PCSG), established in 1975, addressed cracking only in boiling water reactors). One of the objectives of the second PCSG was to include a review of the potential for stress corrosion cracking in Pressurized Water Reactors (PWRs). The results of the study performed by the PCSG were presented in NUREG-0531 (Reference 2-1) entitled "Investigation and Evaluation of Stress-Corrosion Cracking in Piping of Light Water Reactor Plants." In that report the PCSG stated:

> "The PCSG has determined that the potential for stress corrosion cracking in PWR primary system piping is extremely low because the ingredients that produce IGSCC are not all present. The use of hydrazine additives and a hydrogen overpressure limit the oxygen in the coolant to very low levels. Other impurities that might cause stress-corrosion cracking, such as halides or caustic, are also rigidly controlled. Only for brief periods during reactor shutdown when the coolant is exposed to the air and during the subsequent startup are conditions even marginally capable of producing stress-corrosion cracking in the primary systems of PWRs. Operating experience in PWRs supports this determination. To date, no stress corrosion cracking has been reported in the primary piping or safe ends of any PWR."

During 1979, several instances of cracking in PWR feedwater piping led to the establishment of the third PCSG. The investigations of the PCSG reported in NUREG-0691 (Reference 2-2) further confirmed that no occurrences of IGSCC have been reported for PWR primary coolant systems.

The discussion below further qualifies the PCSG's findings.

For stress corrosion cracking (SCC) to occur in piping, the following three conditions must exist simultaneously: high tensile stresses, susceptible material, and a corrosive environment. The potential for stress corrosion is minimized by properly selecting a material immune to SCC as well as preventing of a corrosive environment. The material specifications consider compatibility with the system's operating environment (both internal and external) as well as other material in the system, applicable American Society of Mechanical Engineers (ASME) Code rules, fracture toughness, welding, fabrication, and processing.

The elements of a water environment known to increase the susceptibility of austenitic stainless steel to stress corrosion are oxygen, fluorides, chlorides, hydroxides, hydrogen peroxide, and reduced forms of sulfur (e.g., sulfides, sulfites, and thionates). Strict pipe cleaning standards prior to operation and careful control of water chemistry during plant operation are used to prevent the occurrence of a corrosive environment. Prior to being put into service, the piping is cleaned internally and externally. During flushes and preoperational testing, water chemistry is controlled in accordance with written specifications. Requirements on chlorides, fluorides, conductivity, and pH are included in the acceptance criteria for the piping.

During plant operation, the reactor coolant water chemistry is monitored and maintained within very specific limits. Contaminant concentrations are kept below the thresholds known to be conducive to stress corrosion cracking with the major water chemistry control standards being included in the plant operating procedures as a condition for plant operation. For example, during normal power operation, oxygen concentration in the RCS is expected to be in the ppb range by controlling charging flow chemistry and maintaining hydrogen in the reactor coolant at specified concentrations. Halogen concentrations are also stringently controlled by maintaining concentrations of chlorides and fluorides within the specified limits. Thus, during plant operation, the likelihood of stress corrosion cracking is minimized.

It should be noted that the VCSNS Unit 1 reactor coolant system (RCS) primary loop piping contains Alloy 82/182 DM welds which are susceptible to PWSCC. For the loop A piping, the hot leg spool piece was removed and replaced. In this process, the PWSCC susceptible was replaced with non-susceptible welding material. For loops B and C, the MSIP process was applied at the susceptible hot leg weld locations at the RPVON to modify the through-wall stress profile. This permanent application of MSIP serves to mitigate the effects of PWSCC. VCSNS Unit 1 RCS piping includes Alloy 82 DM welds at the SG nozzle safe-ends. As a protective barrier against PWSCC, Alloy 152 inlay were installed on the inside surface of the Alloy 82 DM nozzle safe-end welds. However, the conservative LBB evaluation of the Alloy 82 DM weld is performed without considering the Alloy 152 weld inlay application at these locations. The LBB evaluation for unmitigated weld locations at the RPVINs includes methodology to address the Alloy 82/182 PWSCC concerns for 80- year plant life in the SLR program. A detailed evaluation of Alloy 82/182 welds is documented in Sections 6.0, 7.4 and 9.0.

2.2 WATER HAMMER

Overall, there is a low potential for water hammer in the RCS since it is designed and operated to preclude the voiding condition in normally filled lines. The reactor coolant system, including piping and primary components, is designed for normal, upset, emergency, and faulted condition transients. The design requirements are conservative relative to both the number of transients and their severity. Relief valve actuation and the associated hydraulic transients following valve opening are considered in the system design. Other valve and pump actuations are relatively slow transients with no significant effect on the system dynamic loads. To ensure dynamic system stability, reactor coolant parameters are stringently controlled. Temperature during normal operation is maintained within a narrow range by control rod position; pressure is controlled by pressurizer heaters and pressurizer spray also within a narrow range for steady-state conditions. The flow characteristics of the system remain constant during a fuel cycle because the only governing parameters, namely system resistance and the reactor coolant pump characteristics, are controlled in the design process. Additionally, Westinghouse has instrumented typical reactor coolant systems to verify the flow and vibration characteristics of the system. Preoperational testing and operating experience have verified the Westinghouse approach. The operating transients of the RCS primary piping are such that no significant water hammer can occur.

2.3 LOW CYCLE AND HIGH CYCLE FATIGUE

Low cycle fatigue considerations are accounted for in the design of the piping system through the fatigue usage factor evaluation to show compliance with the rules of Section III of the ASME Code. A further evaluation of the low cycle fatigue loadings was carried out as part of this study in the form of a fatigue crack growth analysis, as discussed in Section 8.0.

High cycle fatigue loads in the system would result primarily from pump vibrations. These are minimized by restrictions placed on shaft vibrations during hot functional testing and operation. During operation, an alarm signals the exceedance of the vibration limits. Field measurements have been made on a number of plants during hot functional testing, including plants similar to VCSNS Unit 1. Stresses in the elbow below the reactor coolant pump resulting from system vibration have been found to be very small, between 2 and 3 ksi at the highest. These stresses are well below the fatigue endurance limit for the material and would also result in an applied stress intensity factor below the threshold for fatigue crack growth.

2.4 REFERENCES

2-1 Investigation and Evaluation of Stress-Corrosion Cracking in Piping of Light Water Reactor Plants, NUREG-0531, U.S. Nuclear Regulatory Commission, February 1979.

2-2 Investigation and Evaluation of Cracking Incidents in Piping in Pressurized Water Reactors, NUREG-0691, U.S. Nuclear Regulatory Commission, September 1980.

3.0 PIPE GEOMETRY AND LOADING

3.1 INTRODUCTION TO METHODOLOGY

The general approach is discussed first. As an example, a segment of the primary coolant hot leg pipe is shown in Figure 3-1. The as-built outside diameter and minimum wall thickness of the pipe are 33.90 in. and 2.345 in., respectively, as shown in the figure. The normal stresses at the weld locations are from the load combination procedure discussed in Section 3.3 whereas the faulted loads are developed as described in Section 3.4. The components for normal loads are pressure, dead weight, and thermal expansion. For loops B and C, the MSIP loads are considered as a part of the thermal expansion loads. An additional component, safe shutdown earthquake (SSE), is considered for faulted loads. As seen from Table 3-2, the highest faulted stress in all three loops is at the reactor vessel outlet nozzle to pipe weld, location 1. This highest stressed location is a load critical location and is one of the locations at which, as an enveloping location, Leak-Before-Break is to be established. Essentially a circumferential flaw is postulated to exist at this location which is subjected to both the normal loads and faulted loads to assess leakage and stability, respectively. The loads (developed below) at this location are also given in Figure 3-1.

Since the elbows are made of different materials than the pipe, locations other than the highest stressed pipe location are examined taking into consideration both fracture toughness and stress. The elbows are cast stainless steel, and therefore thermal aging must be considered (see Section 4.0). Thermal aging of cast stainless steel (CASS) material results in lower fracture toughness; thus, locations other than the load critical locations must be examined taking into consideration both fracture toughness and stress. The enveloping locations as determined are called toughness critical locations. The most critical locations are apparent only after the full analysis is completed. Once loads (this section) and fracture toughness values (see Section 4.0) are available, the load critical and toughness critical locations are determined (see Section 5.0). At these locations, leak rate evaluations (see Section 6.0) and fracture mechanics evaluations (see Section 7.0) are performed per the guidance of References 3-1 and 3-2. Fatigue crack growth (see Section 8.0) and stability margins are also evaluated (see Section 9.0). All the weld locations considered for the LBB evaluation are those shown in Figure 3-2.

3.2 CALCULATION OF LOADS AND STRESSES

The stresses due to axial loads and total moments are calculated by the following equation:

$$\sigma = \frac{F}{A} + \frac{M}{Z} \tag{3-1}$$

Where:

σ	=	stress, ksi
F	=	axial load, kips
M	=	total moment, in-kips
A	=	pipe cross-sectional area, in^2
Z	=	section modulus, in^3

The total moments for the desired loading combinations are calculated by the following equation:

$$M = \sqrt{M_x^2 + M_y^2 + M_z^2} \tag{3-2}$$

Where,

M	=	total moment for required loading
M_X	=	X component of bending moment
M_Y	=	Y component of bending moment
M_Z	=	Z component of bending moment

The axial load and total moment for leak rate predictions and crack stability analyses are computed by the methods to be explained in Sections 3.3 and 3.4.

3.3 LOADS FOR LEAK RATE EVALUATION

The normal operating loads for leak rate predictions are calculated by the following equations:

$$F = F_{DW} + F_{TH} + F_P \tag{3-3}$$

$$M_X = (M_X)_{DW} + (M_X)_{TH} \tag{3-4}$$

$$M_Y = (M_Y)_{DW} + (M_Y)_{TH} \tag{3-5}$$

$$M_Z = (M_Z)_{DW} + (M_Z)_{TH} \tag{3-6}$$

The subscripts of the above equations represent the following loading cases:

DW	=	deadweight
TH	=	normal thermal expansion. Note that for loops B and C, the MSIP loads are included as a part of the normal thermal expansion loads.
P	=	load due to internal pressure

This method of combining loads is often referred to as the algebraic sum method (References 3-1 and 3-2).

The loads based on this method of combination are provided in Table 3-1 at all the weld locations identified in Figure 3-2. The as-built dimensions are also given in Table 3-1.

3.4 LOAD COMBINATION FOR CRACK STABILITY ANALYSES

In accordance with Standard Review Plan 3.6.3 (References 3-1 and 3-2), the margin in terms of applied loads needs to be demonstrated by crack stability analysis. Margin on loads of 1.4 can be demonstrated if normal plus Safe Shutdown Earthquake (SSE) are applied algebraically and increased by 1.4. The 1.4 margin can be reduced to 1.0 if the deadweight, thermal expansion, internal pressure, Safe Shutdown Earthquake (SSE) inertia and seismic anchor motion (SAM) loads are combined based on individual absolute values as shown in the following equations:

$$F = |F_{DW}| + |F_{TH}| + |F_{P}| + |F_{SSEINERTIA}| + |F_{SSEAM}| \quad (3\text{-}7)$$

$$M_X = |(M_X)_{DW}| + |(M_X)_{TH}| + |(M_X)_{SSEINERTIA}| + |(M_X)_{SSEAM}| \quad (3\text{-}8)$$

$$M_Y = |(M_Y)_{DW}| + |(M_Y)_{TH}| + |(M_Y)_{SSEINERTIA}| + |(M_Y)_{SSEAM}| \quad (3\text{-}9)$$

$$M_Z = |(M_Z)_{DW}| + |(M_Z)_{TH}| + |(M_Z)_{SSEINERTIA}| + |(M_Z)_{SSEAM}| \quad (3\text{-}10)$$

Where subscript SSEINERTIA refers to safe shutdown earthquake inertia and SSEAM is safe shutdown earthquake anchor motion.

The loads so determined are used in the fracture mechanics evaluations (Section 7.0) to demonstrate the LBB margins at the locations established to be the governing locations. These loads at all the weld locations (see Figure 3-2) are given in Table 3-2.

3.5 REFERENCES

3-1 Standard Review Plan: Public Comments Solicited; 3.6.3 Leak-Before-Break Evaluation Procedures; Federal Register/Vol. 52, No. 167/Friday, August 28, 1987/Notices, pp. 32626-32633.

3-2 NUREG-0800 Revision 1, March 2007, Standard Review Plan: 3.6.3 Leak-Before-Break Evaluation Procedures.

Table 3-1
Dimensions, Normal Loads and Normal Stresses for VCSNS Unit 1

Location[a]	Outside Diameter (in.)	Minimum Thickness (in.)	RCS Loop A			RCS Loops B and C		
			Axial Load[b] (kips)	Total Moment (in-kips)	Total Stress (ksi)	Axial Load[b] (kips)	Total Moment (in-kips)	Total Stress (ksi)
1	33.90[c]	2.345[c]	1485	13548	14.28	1339	14213	14.05
2	33.90	2.345	1485	7166	10.56	1339	8520	10.73
3	36.23	2.510	1615	11943	11.77	1513	11068	10.97
4	36.23	2.510	1708	4096	8.38	1657	5757	8.98
5	36.20	2.495	1704	3920	8.33	1651	4418	8.37
6	36.20	2.495	1698	3433	8.08	1646	4192	8.24
7	36.20	2.495	1715	2875	7.87	1710	2131	7.50
8	36.20	2.495	1715	2379	7.63	1710	3173	8.00
9	37.56	3.178	1747	1568	5.66	1799	6511	7.63
10	32.14	2.215	1348	7355	11.52	1380	4371	9.62
11	32.14	2.215	1348	4752	9.73	1380	3634	9.12
12	32.18	2.238	1347	5946	10.55	1373	4530	9.59

Notes:
[a] See Figure 3-2 for piping layout and weld locations.
[b] Axial force includes pressure.
[c] Outside diameter and thickness at location 1 for the hot leg loop A for Inconel 152 are 33.89 inches and 2.40 inches, respectively.

Table 3-2
Dimensions, Faulted Loads and Faulted Stresses for VCSNS Unit 1

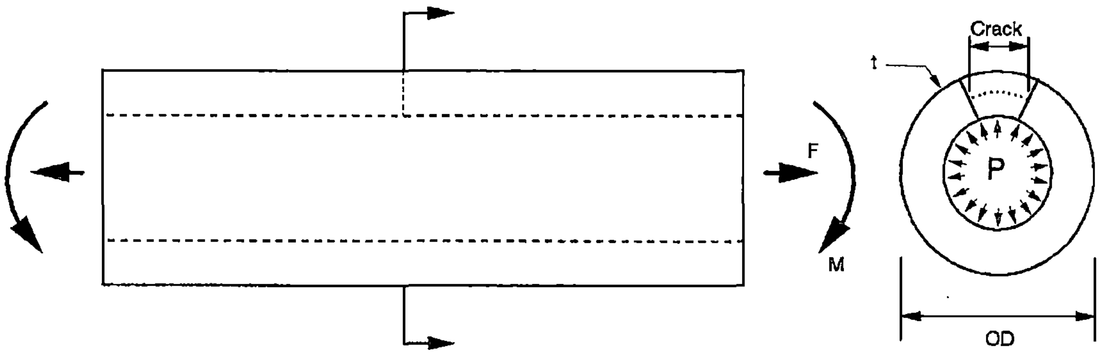
Location[a]	Outside Diameter (in.)	Minimum Thickness (in.)	RCS Loop A			RCS Loops B and C		
			Axial Load[b] (kips)	Total Moment (in-kips)	Total Stress (ksi)	Axial Load[b] (kips)	Total Moment (in-kips)	Total Stress (ksi)
1	33.90[c]	2.345[c]	1963	23194	21.96	2108	23845	22.96
2	33.90	2.345	1977	21927	21.28	2123	23288	22.70
3	36.23	2.510	1948	31524	22.35	2049	30731	22.35
4	36.23	2.510	1789	16741	14.71	1840	18784	15.88
5	36.20	2.495	1756	12234	12.52	1808	14040	13.58
6	36.20	2.495	1773	7440	10.28	1825	9502	11.47
7	36.20	2.495	1814	8446	10.92	1809	7027	10.22
8	36.20	2.495	1809	8212	10.79	1805	9202	11.25
9	37.56	3.178	1799	10813	9.21	1851	15682	11.15
10	32.14	2.215	1491	11753	15.22	1523	9332	13.71
11	32.14	2.215	1512	9300	13.64	1544	8405	13.18
12	32.18	2.238	1481	8530	12.96	1505	7953	12.54

Notes:

[a] See Figure 3-2 for piping layout and weld locations.

[b] Axial force includes pressure.

[c] Outside diameter and thickness at location 1 for the hot leg loop A for Inconel 152 are 33.89 inches and 2.40 inches, respectively.



Location 1

OD[a,d] = 33.90 in
t[a,d] = 2.345 in

Normal Loads[a]

Axial Force[c]:

Loop A – 1485 kips

Loop B/C – 1339 kips

Total Moment:

Loop A – 13548 in-kips

Loop B/C – 14213 in-kips

Faulted Loads[b]

Axial Force[c]:

Loop A – 1963 kips

Loop B/C – 2108 kips

Total Moment:

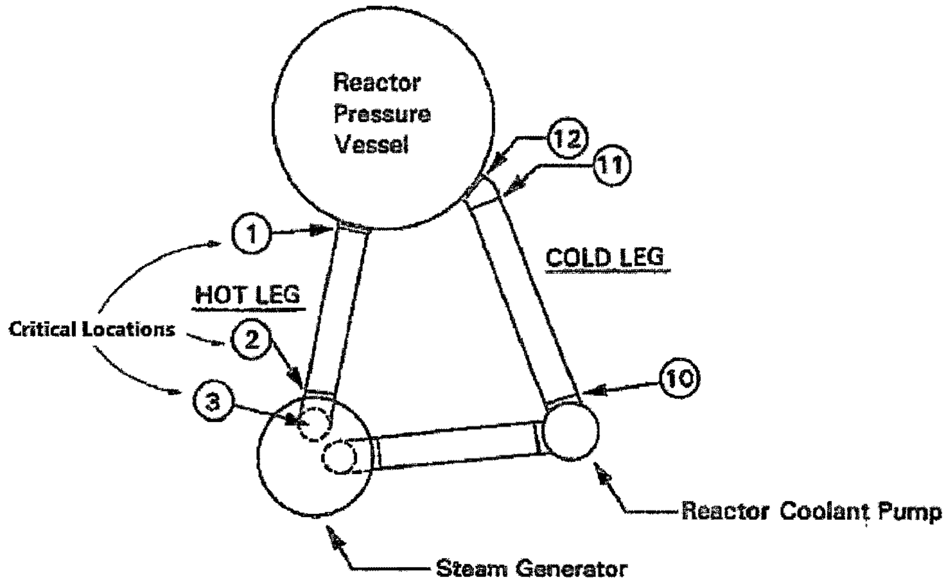
Loop A – 23194 in-kips

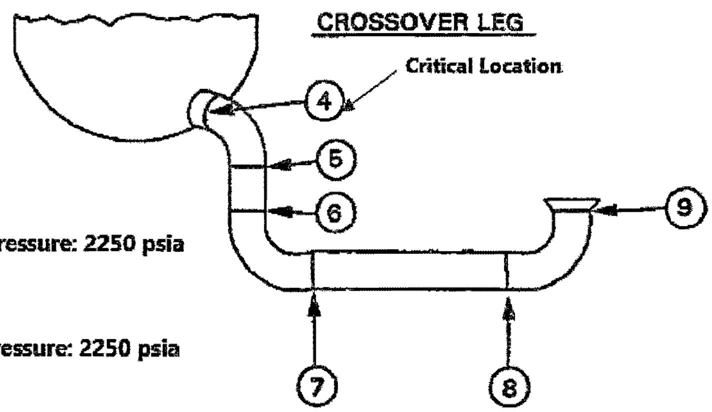
Loop B/C – 23845 in-kips

Notes:

[a] See Table 3-1.
[b] See Table 3-2.
[c] Includes the force due to a pressure of 2250 psia.
[d] See Note c on Table 3-1.

Figure 3-1
Hot Leg Coolant Pipe





HOT LEG

Temperature 622°F, Pressure: 2250 psia

CROSSOVER LEG

Temperature 553°F, Pressure: 2250 psia

COLD LEG

Temperature 553°F, Pressure: 2250 psia

Figure 3-2
Schematic Diagram of VCSNS Unit 1 Primary Loop Showing Weld Locations

4.0 MATERIAL CHARACTERIZATION

4.1 PRIMARY LOOP PIPE AND FITTINGS MATERIALS

The primary loop pipe materials are SA376 TP304N and the elbow fittings are SA351 CF8A.

4.2 TENSILE PROPERTIES

The Certified Materials Test Reports (CMTRs) for VCSNS Unit 1 were used to establish the tensile properties for the Leak-Before-Break analyses. The CMTRs include tensile properties at room temperature for each of the heats of material. These properties for VCSNS Unit 1 primary loop piping and fittings are given in Tables 4-1 and 4-2, and for Inconel 152 weld in Table 4-3. The average yield strength and the minimum yield and ultimate strengths are also identified.

Piping: For the SA376 TP304N material, the properties at operating temperatures were established from the tensile properties at room temperature given in Table 4-1 by utilizing Section II of the ASME Boiler and Pressure Vessel (B&PVC) Code 2007 Edition up to and including 2008 Addenda (Reference 4-1). Code tensile properties at the normal operating temperatures (622°F for Hot Leg and 553°F for Crossover leg and Cold leg) were obtained by a linear interpolation of the tensile properties provided in the Code. The ratios of the tensile properties at the applicable operating temperatures to the corresponding tensile properties at room temperature were then applied to the room temperature values given in Table 4-1 to obtain the plant specific properties for SA376 TP304N at normal operating temperatures. Table 4-1 includes the replacement spool piece tensile properties. As shown in Table 4-1 minimum tensile properties remain unchanged, and the average yield strength is reduced by 0.4% (from 46.7 ksi to 46.5 ksi). The average yield strength of 46.7 ksi at room temperature from Table 4-1 is used to calculate the average yield strength at operating temperature as shown in Table 4-4. The calculated value in Table 4-4 is used in the leakage analysis for the SA376 TP304N pipe material since it results in a larger leakage flaw size.

Fittings (Elbows): For the SA351 CF8A material, the properties at operating temperature were established from the tensile properties at room temperature given in Table 4-2. The representative tensile properties for SA351 CF8A at operating temperatures (622°F for Hot Leg and 553°F for Crossover leg and Cold leg) were obtained by utilizing Section II of the ASME B&PVC Code 2007 Edition up to and including 2008 Addenda (Reference 4-1). Code tensile properties at the applicable operating temperatures considered in this LBB analysis were obtained by a linear interpolation of the tensile properties provided in the Code. To obtain the plant specific properties for SA351 CF8A at operating temperatures of 622°F and 553°F as shown in Table 4-4, the Code minimum properties at the applicable operating temperatures were adjusted to account for the actual yield strength and ultimate tensile strength from the CMTR values at room temperature given in Table 4-2.

Weld Material: Table 4-3 shows the room temperature tensile properties for the Inconel 152 material. The properties at operating temperature were established from the tensile properties at room temperature given in Table 4-3 by utilizing Section II of the ASME B&PVC Code 2007 Edition up to and including 2008 Addenda (Reference 4-1). A similar method, as indicated above, is used to obtain the plant specific properties for Inconel 152 at operating temperature of 622°F for Hot Leg piping presented in Table 4-4.

For Alloy 82/182 DM welds CMTR data was not available, and the typical tensile properties from Westinghouse source for the 82/182 weld material at the applicable operating temperatures as listed in Table 4-4 are used in the LBB evaluation.

The LBB evaluation considers the normal operating temperature of 622°F for Hot leg and 553°F for Crossover and Cold legs for material property interpolation.

The average and lower bound yield strengths and ultimate strengths at operating temperatures of 622°F and 553°F which are used in the LBB evaluation are summarized in Table 4-4. The ASME Code values for modulus of elasticity at the applicable operating temperatures are also provided. Poisson's ratio was taken as 0.3.

It should be noted that there is no significant impact by using the ASME Code Section II edition up to and including 2008 Addenda for material properties for the LBB analysis, as compared to the VCSNS ASME Code of record, i.e.: ASME B&PVC Code Section III, Appendix I, 1971 edition including Addenda through Winter 1971.

4.3 FRACTURE TOUGHNESS PROPERTIES

The pre-service fracture toughness (J) of cast austenitic stainless steel (CASS) that are of interest are in terms of J_{IC} (J at Crack Initiation) and have been found to be very high at 600°F. [

]^{a,c,e} However, cast stainless steel is susceptible to thermal aging at the reactor operating temperature, that is, about 550°F. Thermal aging of cast stainless steel results in embrittlement, which means a decrease in the ductility, impact strength, and fracture toughness of the material. Depending on the material composition, the Charpy impact energy of a cast stainless steel component could decrease to a small fraction of its original value after exposure to reactor temperatures during service.

The susceptibility of the material to thermal aging increases with increasing ferrite and molybdenum contents.

In 1994, the Argonne National Laboratory (ANL) completed an extensive research program in assessing the extent of thermal aging of cast stainless steel materials (Reference 4-2). The ANL research program measured mechanical properties of cast stainless steel materials after they had been heated in controlled ovens for long periods of time. ANL compiled a data base, both from data within ANL and from international sources, of about 85 compositions of cast stainless steel exposed to a temperature range of 290°-400°C (550°-750°F) for up to 58,000 hours (6.5 years). In 2015, the work done by ANL was augmented, and the fracture toughness database for CASS materials was aged to 100,000 hours at 290°–350°C (554°–633°F). The methodology for estimating fracture properties has been extended to cover CASS materials with a ferrite content of up to 40%. From this database (NUREG/CR-4513, Revision 2), ANL developed correlations for estimating the extent of thermal aging of cast stainless steel (Reference 4-4). From this data base (NUREG/CR-4513, Revision 2), ANL developed correlations for estimating the extent of thermal aging of cast stainless steel (Reference 4-4).

ANL developed the fracture toughness estimation procedures by correlating data in the database conservatively. After developing the correlations, ANL validated the estimation procedures by comparing the estimated fracture toughness with the measured value for several cast stainless steel plant components removed from actual plant service. The procedure developed by ANL in Revision 1 and Revision 2 of NUREG/CR-4513 (References 4-3 and 4-4) was used to calculate the end of life limiting fracture toughness values of the CASS elbows for the cold leg, crossover leg and hot leg locations. Note that LBB analyses have acceptable margins when performing the elastic-plastic J-integral evaluations with the use of lower bound fracture toughness properties from NUREG/CR-4513, Revision 1 and Revision 2. Furthermore, this report used saturated toughness approved by NRC in NUREG/CR-4513 Revision 1 and Revision 2. Therefore, the LBB analysis is acceptable from a saturated toughness perspective.

The method described below was used to calculate the toughness properties for the cast material, SA351 CF8A, of the VCSNS Unit 1 primary coolant loop elbows.

The J_{Ic}, J_{max} and T_{mat} values for each material heat for CASS elbows was calculated using both Revision 1 and Revision 2 of NUREG/CR-4513 (References 4-3 and 4-4), and the enveloped values from both revisions are summarized in Table 4-6. While Revision 1 (Reference 4-3) provides more limiting J_{Ic} values for material heats, it was found that Revision 2 (Reference 4-4) resulted in the most limiting fracture toughness (J_{max} and T_{mat}) values for the material heats identified in Table 4-6. Therefore, the J_{Ic} values in Table 4-6 are based on Revision 1 of NUREG/CR-4513, and the reported values for J_{max} and T_{mat} in Table 4-6 are based on Revision 2 of NUREG/CR-4513.

Based on Reference 4-4, the lower bound fully aged fracture toughness correlations are used for the SA351 CF8A material.

The chemical compositions of the VCSNS Unit 1 primary loop elbow fitting material (SA351 CF8A) are available from CMTRs and are provided in Table 4-5. The following equations are taken from References 4-3 and 4-4. Note that equations provided below for both revisions are the same, except for Equations 4-8 and 4-10. Equations 4-8.a and 4.10.a are from Reference 4-3, and Equations 4-8.b and 4.10.b are from Reference 4-4:

$$Cr_{eq} = Cr + 1.21(Mo) + 0.48(Si) - 4.99 = \text{(Chromium equivalent)} \quad (4\text{-}1)$$

$$Ni_{eq} = (Ni) + 0.11(Mn) - 0.0086(Mn)^2 + 18.4(N) + 24.5(C) + 2.77 = \text{(Nickel equivalent)} \quad (4\text{-}2)$$

$$\delta_c = 100.3(Cr_{eq} / Ni_{eq})^2 - 170.72(Cr_{eq} / Ni_{eq}) + 74.22 = \text{(Ferrite Content)} \quad (4\text{-}3)$$

where the elements are in percent weight and δ_c is ferrite in percent volume.

The saturation room temperature (RT) impact energies of the cast stainless steel materials were determined from the chemical compositions available from CMTRs and provided in Table 4-5.

For CF8A steel, the saturation value of RT impact energy Cv_{sat} (J/cm^2) is the lower value determined from:

$$\log_{10} Cv_{sat} = 1.15 + 1.36 \exp(-0.035\phi) \quad (4\text{-}4)$$

Where the material parameter ϕ is expressed as:

$$\phi = \delta_c\,(Cr + Si)(C + 0.4N) \qquad (4\text{-}5)$$

and from:

$$\log_{10}Cv_{sat} = 5.64 - 0.006\delta_c - 0.185Cr + 0.273Mo - 0.204Si + 0.044Ni - 2.12(C + 0.4N) \qquad (4\text{-}6)$$

The saturation J-R curve at RT, for static-cast CF8A steel is given by:

$$J_d = 49\,(Cv_{sat})^{0.52}(\Delta a)^n \qquad (4\text{-}7)$$

$$n = 0.20 + 0.12\,\log10\,(Cv_{sat}) \qquad (4\text{-}8.a)$$

$$n = 0.18 + 0.10\,\log10\,(Cv_{sat}) \qquad (4\text{-}8.b)$$

Where J_d is the "deformation J" in kJ/m^2 and Δa is the crack extension in mm.

The saturation J-R curve at 290°C (554°F), for static-cast CF8A steel is given by:

$$J_d = 102\,(Cv_{sat})^{0.28}(\Delta a)^n \qquad (4\text{-}9)$$

$$n = 0.21 + 0.09\,\log10\,(Cv_{sat}) \qquad (4\text{-}10.a)$$

$$n = 0.17 + 0.09\,\log_{10}\,(Cv_{sat}) \qquad (4\text{-}10.b)$$

Where J_d is the "deformation J" in kJ/m^2 and Δa is the crack extension in mm.

[

]a,c,e

The critical heats with the most limiting allowable fracture values (lowest fracture toughness properties values and lowest tearing modulus) for VCSNS Unit 1 primary loop elbows from Table 4-7 is selected as shown below:

[ ]a,c,e

Toughness properties from these material heats are conservatively used for all the critical location evaluations.

J_{Ic} and J_{max} Calculations:

[

]a,c,e

T_{mat} Calculations:

The material tearing modulus, T_{mat}, is calculated as follows:

$$T_{mat} = dJ/da \times E/(\sigma_{fa})^2$$

The results from the ANL Research Program indicate that the lower-bound fracture toughness of thermally aged cast stainless steel is similar to that of submerged arc welds (SAWs). In addition, historic testing done on representative plants documented in References 4-5 and 4-6, has shown that the wrought and cast stainless steel piping exhibits more limiting (unaged) fracture toughness properties than the weld metal. Since the CASS material's aged lower bound fracture toughness values are similar to that of Submerged Arc Welds (SAWs), and since SAWs are considered to be the most limiting of welding processes (with respect to GTAW and SMAW), it is concluded that the aged fracture toughness of the wrought and cast base metal is more limiting than the aged fracture toughness of the stainless-steel weld metal. Therefore, the stainless-steel weld regions are less limiting than the cast material, and the applied value of the J-integral for a flaw in the weld regions will be lower than that in the base metal because the yield stress for the stainless steel weld materials is much higher at operating temperature[(a)].

Forged stainless steel piping such as SA376 TP304N does not degrade due to thermal aging. Thus, fracture toughness values well in excess of that established for the cast material exist for this material throughout service life and are not limiting.

Inconel 152 and Alloy 82/182 weld materials have high toughness values and do not degrade due to thermal aging. As discussed in Reference 4-7, the fracture resistance of Ni Alloys (Alloys 82 and 52) and their welds have been investigated by conducting fracture toughness J-R curve tests at 24–338°C in deionized water. The results indicated that Alloy 690 welds exhibit excellent fracture toughness in air and high-temperature water (> 93 °C).

Since nickel alloys are known to have high toughness properties and because CF8A CASS base metal is susceptible to thermal aging degradation of the fracture toughness, it is determined that the CF8A CASS base metal presents the most limiting condition. Therefore, in the fracture mechanics analyses that follow, the thermally aged fracture toughness allowables of the CASS material given in Table 4-7 will be used as the criteria against which the calculated applied fracture toughness values will be compared.

4.4 REFERENCE

4-1 ASME Boiler and Pressure Vessel Code Section II, Part D, "Properties (Customary) Materials," 2007 Edition up to and including 2008 Addenda.

4-2 O. K. Chopra and W. J. Shack, "Assessment of Thermal Embrittlement of Cast Stainless Steels," NUREG/CR-6177, U.S. Nuclear Regulatory Commission, Washington, DC, May 1994.

4-3 O. K. Chopra, "Estimation of Fracture Toughness of Cast Stainless Steels During Thermal Aging in LWR Systems," NUREG/CR-4513, Revision 1, U.S. Nuclear Regulatory Commission, Washington, DC, August 1994.

4-4 O. K. Chopra, "Estimation of Fracture Toughness of Cast Stainless Steels During Thermal Aging in LWR Systems," NUREG/CR-4513, Revision 2, U.S. Nuclear Regulatory Commission, Washington, DC, May 2016" including Errata, March 15, 2021.

4-5 Westinghouse Report, WCAP-9787, "Tensile and Toughness Properties of Primary Piping Weld Metal for Use in Mechanistic Fracture Evaluation," May 1981.

4-6 Westinghouse Report, WCAP-9558, Revision 2, "Mechanistic Fracture Evaluation of Reactor Coolant Pipe Containing a Postulated Circumferential Through-Wall Crack," May 1981.

4-7 NUREG/CR-6721, "Effects of Alloy Chemistry, Cold Work, and Water Chemistry on Corrosion Fatigue and Stress Corrosion Cracking of Nickel Alloys and Welds," April 2001.

[a] In the report, all the applied J values were conservatively determined by using base metal strength properties.

Table 4-1
Measured Room Temperature Tensile Properties for VCSNS Unit 1 Primary Loop Piping

Component	Loop	Heat Number	Material Type	Room Temperature Properties (ksi)	
				Yield Strength	Ultimate Strength
Hot Leg	A	L1359/14462	SA376 TP304N	48.0 49.7	87.7 93.7
Hot Leg	B	L1359/14463	SA376 TP304N	47.2 49.7	88.4 92.4
Hot Leg	C	L1359/14464	SA376 TP304N	45.9 49.7	88.7 91.7
Crossover Leg	A	K3723/15903X	SA376 TP304N	46.5 48.2	89.7 91.4
Crossover Leg	A	K3723/15692W	SA376-TP304N	45.0 48.0	87.4 91.4
Crossover Leg	B	K3723/15903Y	SA376 TP304N	46.5 48.2	89.7 91.4
Crossover Leg	B	K3723/15692X	SA376 TP304N	45.0 48.0	87.4 91.4
Crossover Leg	C	K3723/15903Z	SA376 TP304N	46.5 48.2	89.7 91.4
Crossover Leg	C	K3723/15692Y	SA376 TP304N	45.0 48.0	87.4 91.4
Cold Leg	A	L1336/14461	SA376 TP304N	40.9 45.9	82.1 88.6
Cold Leg	B	K3723/16098	SA376 TP304N	43.7 45.9	86.4 89.9
Cold Leg	C	L1551/17989	SA376 TP304N	48.5 43.7	92.2 86.1
Hot Leg Spool Piece	A	J6347	SA376 TP304N	41.2	83.6
Average				46.7 (46.5*)	

Room Temp. Minimum Yield Strength: 40.9 ksi
Room Temp. Minimum Ultimate Strength: 82.1 ksi
Room Temp. Average Yield Strength: 46.7 (*46.5, with spool piece) ksi

Table 4-2
Measured Room Temperature Tensile Properties for VCSNS Unit 1 Primary Loop Fittings (Elbows)

Component	Loop	Heat Number	Material Type	Room Temperature Properties (ksi)	
				Yield Strength	Ultimate Strength
Hot Leg	A	79420-1	SA351 CF8A	38.1	81.5
Hot Leg	B	79420-2	SA351 CF8A	38.1	81.5
Hot Leg	C	80019-2	SA351 CF8A	35.2	77.7
Crossover Leg	A	90946-1	SA351 CF8A	38.7	86.7
Crossover Leg	A	89916-1	SA351 CF8A	44.3	86.9
Crossover Leg	B	92293-1	SA351 CF8A	38.2	82.2
Crossover Leg	B	89253-1	SA351 CF8A	42.5	85.7
Crossover Leg	C	91068-1	SA351 CF8A	37.3	82.5
Crossover Leg	C	93212-1	SA351 CF8A	37.9	83.7
Crossover Leg	B	84013-1	SA351 CF8A	39.3	83.5
Crossover Leg	C	84227-1	SA351 CF8A	40.5	82.5
Crossover Leg	A	82695-2	SA351 CF8A	37.2	82.4
Cold Leg	A	67487-1	SA351 CF8A	43.6	86.5
Cold Leg	B	65692-2	SA351 CF8A	39.3	83.6
Cold Leg	C	73359-4	SA351 CF8A	37.0	85.9
Average				39.2	

Room Temp. Minimum Yield Strength: 35.2 ksi
Room Temp. Minimum Ultimate Strength: 77.7 ksi
Room Temp. Average Yield Strength: 39.2 ksi

Table 4-3 Measured Room Temperature Tensile Properties for Hot Leg Loop A Inconel 152 Weld at Location 1					
Component	**Loop**	**Heat Number**	**Material Type**	**Room Temperature Properties (ksi)**	
				Yield Strength	**Ultimate Strength**
Hot Leg	A	45D2	Inconel 152	61.60	101.20
Hot Leg	A	WC59D8	Inconel 152	59.60	99.50
Hot Leg	A	49D4	Inconel 152	62.70	101.20
Average				61.30	

Room Temp. Minimum Yield Strength: 59.60 ksi
Room Temp. Minimum Ultimate Strength: 99.50 ksi
Room Temp. Average Yield Strength: 61.30 ksi

Table 4-4
Mechanical Properties for VCSNS Unit 1 Materials at Operating Temperatures

a,c,e

Modulus of Elasticity at operating temperatures

For SA376 TP304N and SA351 CF8A: E = 25.19 x 10^6 psi at T=622°F; E = 25.54 x 10^6 psi at T=553°F
For Inconel 152: E = 27.99 x 10^6 psi at T=622°F
For Alloy 82/182: E = 28.61 x 10^6 psi at T=622°F; E = 28.84 x 10^6 psi at T=553°F
Poisson's ratio: 0.3

Table 4-5
Chemistry and Fracture Toughness Properties
of the SA351 CF8A Material Heats of VCSNS Unit 1

a,c,e

Table 4-6
Enveloped J_{Ic}, J_{max}, T_{mat} for the SA351 CF8A Material Heats from Revision 1 and Revision 2 of NUREG/CR-4513

a,c,e

Table 4-7
Fracture Toughness Properties of SA351 CF8A for VCSNS Unit 1 Primary Loops for Leak-Before-Break Evaluation at Critical Locations

a,c,e

a,c,e

Figure 4-1
Pre-Service J vs. Δa for SA351-CF8M Cast Stainless Steel at 600°F

Note: This plot is shown for demonstration purposes. While the material relevant to this evaluation is SA351 CF8A, the toughness characteristic of SA351 CF8M, shown here, are also for reference only.

5.0 CRITICAL LOCATION AND EVALUATION CRITERIA

5.1 CRITICAL LOCATIONS

The Leak-Before-Break (LBB) evaluation margins are to be demonstrated for the limiting locations (governing locations). Candidate locations are designated load critical locations or toughness critical locations as discussed in Section 3.0. Such locations are established considering the loads (Section 3.0) and the material properties established in Section 4.0. Separate critical locations are created for VCSNS RCS primary loop A and loops B&C piping as described below. The critical locations are shown in Figure 3-2.

Load Critical Locations

The highest stressed location for the SA376 TP304N straight pipes is location 1 for each of the three loops. For the weaker SA351 CF8A elbows: the highest faulted stressed welds at Hot leg are locations 2 and 3 for loops A, B and C; the highest faulted stress location at Crossover leg (XL) and Cold leg (CL) for each of the three loops is at location 4.

For the Alloy 82/182 DM welds at the SGIN, SGON and RPVIN to safe-end, the highest faulted stress locations are at the SGIN to safe-end, location 3, for loops A, B and C. Location 4 is selected as a representative location for evaluation of the Alloy 82/182 DM welds at the XL and the CL (location 4 and location 12). Evaluation of Alloy 82/182 DM welds at the RPVON location 1, loops B and C, is performed as well.

Toughness Critical Location

Low toughness locations are at the end of each elbow since the elbows are made of cast materials and can be susceptible to thermal aging. Per Section 4.3, the critical material location for the elbows is [ ]^{a,c,e}, due to low toughness. As identified above the highest faulted stresses at the elbow locations are 2, 3, 4 and 12 (location 4 is the governing location for both locations 4 and 12) for each of the three loops. The limiting toughness value determined in Section 4.3 is conservatively used in evaluating of these locations.

For the critical locations, the tensile properties are shown in Table 4-4, and the allowable toughness properties are shown in Table 4-7.

5.2 FRACTURE CRITERIA

As will be discussed later, fracture mechanics analyses are made based on loads and postulated flaw sizes related to leakage. The stability criteria against which the calculated J (i.e., J_{app}) and tearing modulus (T_{app}) are compared are:

(1) If $J_{app} < J_{IC}$, then an existing crack is stable (or a crack will not initiate);

(2) If $J_{app} \geq J_{IC}$; and $T_{app} < T_{mat}$ and $J_{app} < J_{max}$, then the crack is stable.

Where:
J_{app} = Applied J
J_{IC} = J at Crack Initiation
T_{app} = Applied Tearing Modulus

T_{mat} = Material Tearing Modulus
J_{max} = Maximum J value of the material

For critical locations, the limit load method discussed in Section 7.0 was also used.

For global failure mechanism, the stability analysis is performed using limit load method based on loads and postulated flaw sizes related to leakage, with the criteria as follows:

- Margin of 10 on the Leak Rate
- Margin of 2.0 on Flaw Size
- Margin of 1.0 on Loads (using the absolute summation method for faulted load combination).

6.0 LEAK RATE PREDICTIONS

6.1 INTRODUCTION

The purpose of this section is to discuss the method which is used to predict the flow through postulated through-wall cracks and present the leak rate calculation results for through-wall circumferential cracks.

6.2 GENERAL CONSIDERATIONS

The flow of hot pressurized water through an opening to a lower back pressure causes flashing which can result in choking. For long channels where the ratio of the channel length, L, to hydraulic diameter, D_H, (L/D_H) is greater than [

]^{a,c,e}

6.3 CALCULATION METHOD

The basic method used in the leak rate calculations is the method developed by [

]^{a,c,e}

The flow rate through a crack was calculated in the following manner. Figure 6-1 (from Reference 6-2) was used to estimate the critical pressure, P_c, for the primary loop enthalpy condition and an assumed flow. Once P_c was found for a given mass flow, the [

]^{a,c,e} was found from Figure 6-2 (taken from Reference 6-2). For all cases considered, since [

].^{a,c,e} Therefore, this method will yield the two-phase pressure drop due to momentum effects as illustrated in Figure 6-3, where P_o is the operating pressure. Now using the assumed flow rate, G, the frictional pressure drop can be calculated using:

$$\Delta P_f = \left[\qquad\qquad \right]^{a,c,e} \tag{6-1}$$

where the friction factor f is determined using the [ ]^{a,c,e} The crack relative roughness, ε, was obtained from fatigue crack data on stainless steel samples. The relative roughness value used in these calculations was [ ]^{a,c,e}

The frictional pressure drop using Equation 6-1 is then calculated for the assumed flow rate and added to the [ ]^{a,c,e} to obtain the total pressure drop from the primary system to the atmosphere.

That is, for the primary loop:

Absolute Pressure - 14.7 = [ ]^{a,c,e} (6-2)

for a given assumed flow rate G. If the right-hand side of Equation 6-2 does not agree with the pressure difference between the primary loop and the atmosphere, then the procedure is repeated until Equation 6-2 is satisfied to within an acceptable tolerance which in turn leads to flow rate value for a given crack size.

6.4 LEAK RATE CALCULATIONS

Leak rate calculations were made as a function of crack length at the governing locations previously identified in Section 5.1. The normal operating loads of Table 3-1 were applied in these calculations. The crack opening areas were estimated using the method of Reference 6-3, and the leak rates were calculated using the two-phase flow formulation described above. The average material properties of Section 4.0 (see Table 4-4) were used for these calculations.

The flaw sizes to yield a leak rate of 10 gpm for VCSNS Unit 1 were calculated at the governing locations with pipe material SA376-TP304N, elbow material SA351-CF8A, and weld material Alloy 82/182 are given in Table 6-1, Table 6-2 and Table 6-3, respectively. Table 6-3 also shows the Inconel 152 weld flaw size for a 10 gpm leak rate for location 1 of the loop A hot leg. The flaw sizes, so determined, are called leakage flaw sizes. Based on the PWSCC crack morphology, an increase factor of 1.69 between the PWSCC and fatigue crack morphologies (Reference 6-4) is applied to the leakage flaw sizes for the Alloy 82/182 DM welds as shown in Table 6-3.

The VCSNS Unit 1 RCS pressure boundary leak detection system meets the intent of Regulatory Guide 1.45 (Reference 6-5), and the plant leak detection capability is 1 gpm. Thus, to satisfy the margin of 10 on the leak rate, the flaw sizes (leakage flaw sizes) are determined which yield a leak rate of 10 gpm.

6.5 REFERENCES

6-1 [ ]^{a,c,e}

6-2 M. M, El-Wakil, "Nuclear Heat Transport, International Textbook Company," New York, N.Y., 1971.

6-3 Tada, H., "The Effects of Shell Corrections on Stress Intensity Factors and the Crack Opening Area of Circumferential and a Longitudinal Through-Crack in a Pipe," Section II-1, NUREG/CR-3464, September 1983.

6-4 D. Rudland, R. Wolterman, G. Wilkowski, R. Tregoning, "Impact of PWSCC and Current Leak Detection on Leak-Before-Break," proceedings of Conference on Vessel Head Penetration, Inspection, Cracking, and Repairs, Sponsored by USNRC, Marriot Washingtonian Center, Gaithersburg, MD, September 29 to October 2, 2003.

6-5 Regulator Guide 1.45, Revision 1, "Guidance on Monitoring and Responding to Reactor Coolant System Leakage," 2008.

Table 6-1
Flaw Sizes Yielding a Leak Rate of 10 gpm
at the Critical Locations with SA376-TP304N Material

a,c,e

Table 6-2
Flaw Sizes Yielding a Leak Rate of 10 gpm
at the Elbow Critical Locations
with SA 351-CF8A Material

a,c,e

Table 6-3
Flaw Sizes Yielding a Leak Rate of 10 gpm
at the Critical Locations with Alloy 82/182 and Inconel 152 Materials

a,c,e

a,c,e

MASS VELOCITY (lb/(sec)(ft^2))

STAGNATION ENTHALPY (10^2 Btu/lb)

Figure 6-1
Analytical Predictions of Critical Flow Rates
of Steam-Water Mixtures

a,c,e

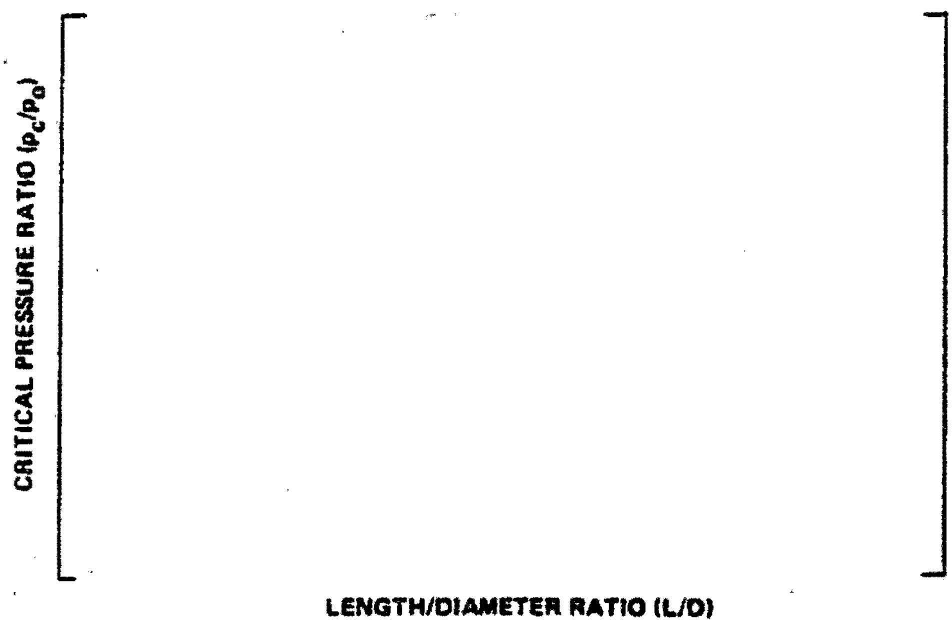


Figure 6-2

[ ]a,c,e Pressure Ratio as a Function of L/D

a,c,e

[ ]

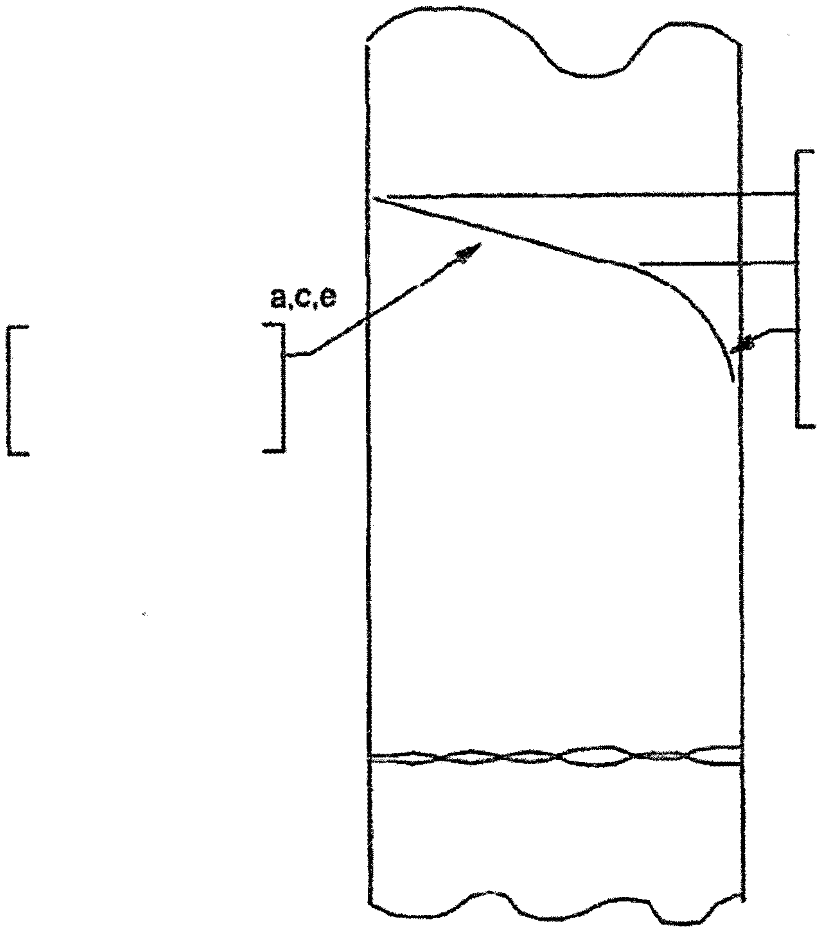


Figure 6-3
Idealized Pressure Drop Profile Through a Postulated Crack

7.0 FRACTURE MECHANICS EVALUATION

7.1 LOCAL FAILURE MECHANISM

The local mechanism of failure is primarily dominated by the crack tip behavior in terms of crack-tip blunting, initiation, extension, and final crack instability. The local stability will be assumed if the crack does not initiate at all. It has been accepted that the initiation toughness measured in terms of J_{IC} from a J-integral resistance curve is a material parameter defining the crack initiation. If, for a given load, the calculated J-integral value is shown to be less than the J_{IC} of the material, then the crack will not initiate. If the initiation criterion is not met, one can calculate the tearing modulus as defined by the following relation:

$$T_{app} = \frac{dJ}{da} x \frac{E}{\sigma_f^2} \tag{7-1}$$

Where:

T_{app}	=	applied tearing modulus
E	=	modulus of elasticity
σ_f	=	$0.5\ (\sigma_y + \sigma_u)$ = flow stress
a	=	crack length
σ_y, σ_u	=	yield and ultimate strength of the material, respectively

Stability is said to exist when ductile tearing does not occur if T_{app} is less than T_{mat}, the experimentally determined tearing modulus. Since a constant T_{mat} is assumed, a further restriction is placed in J_{app}. J_{app} must be less than J_{max}; where J_{max} is the maximum value of J for which the experimental T_{mat} is greater than or equal to the T_{app} used.

As discussed in Section 5.2 the local crack stability criteria is a two-step process:

(1) If $J_{app} < J_{IC,}$ then an existing crack is stable (or a crack will not initiate);

(2) If $J_{app} \geq J_{IC}$; and $T_{app} < T_{mat}$ and $J_{app} < J_{max}$, then the crack is stable.

7.2 GLOBAL FAILURE MECHANISM

Determination of the conditions which lead to failure in stainless steel should be done with plastic fracture methodology because of the large amount of deformation accompanying fracture. One method for predicting the failure of ductile material is the plastic instability method, based on traditional plastic limit load concepts, but accounting for strain hardening and taking into account the presence of a flaw. The flawed pipe is predicted to fail when the remaining net section reaches a stress level at which a plastic hinge is formed. The stress level at which this occurs is termed as the flow stress. The flow stress is generally taken as the average of the yield and ultimate tensile strength of the material at the temperature of interest. This methodology has been shown to be applicable to ductile piping through a large number of experiments and will be used here to predict the critical flaw size in the primary coolant piping. The failure criterion has been obtained by requiring equilibrium of the section containing the flaw (Figure 7-1) when loads are applied. The detailed development is provided in Appendix A for a through-wall circumferential flaw in a pipe with internal pressure, axial force, and imposed bending moments. The limit moment for such a pipe is given by:

[ ]^{a,c,e}

Where:

[

]^{a,c,e}

The analytical model described above accurately accounts for the piping internal pressure as well as imposed axial force as they affect the limit moment. Good agreement was found between the analytical predictions and the experimental results (Reference 7-1).

For application of the limit load methodology, the material, including consideration of the configuration, must have a sufficient ductility and ductile tearing resistance to sustain the limit load.

7.3 RESULTS OF CRACK STABILITY EVALUATION

As discussed in Sections 7.1 and 7.2, the LBB evaluation for VCSNS Unit 1 consists of evaluating two failure mechanisms. Stability analyses were performed at the critical locations established in Section 5.1.

The elastic-plastic fracture mechanics (EPFM) J-integral analyses for through-wall circumferential cracks in a cylinder were performed using the procedure in the EPRI fracture mechanics handbook (Reference 7-2).

The more limiting lower-bound tensile properties for base metal for SA351-CF8A elbow material from Section 4.0 were applied (see Table 4-4). The fracture toughness properties established in Section 4.3, and the normal plus SSE loads given in Table 3-2 were used for the EPFM calculations. Evaluations were performed at the toughness critical locations identified in Section 5.1. Note that one bounding J-integral evaluation was performed for both locations 2 and 3 of loop A and loops B&C using the most limiting geometry, the highest faulted loads from both locations as provided in Table 3-2. Also, the highest leakage flaw size from both locations 2 and 3 is used in the J-integral evaluation. The results of the elastic-plastic fracture mechanics J-integral evaluations are given in Table 7-1. The associated leakage flaw sizes from Table 6-2 are also presented in Table 7-1, except for locations 2 and 3, where the highest from both leakage flaw sizes for locations 2 and 3 of loop A and loops B&C in Table 6-2 is used in the J-integral evaluation. In addition, one bounding J-integral evaluation was performed at governing location 4 for both locations 4 and 12 for all three loops, and results are presented in Table 7-1.

A stability analysis based on limit load as described in Section 7.2 was performed for critical locations 1, 2, 3 and 4 (location 4 is the governing location for both locations 4 and 12). Table 7-2 and Table 7-3 summarize the results of the stability analyses based on the limit load for piping SA376 TP304N and SA351-CF8A elbow materials, respectively. The associated leakage flaw sizes (from Table 6-1 and Table 6-2) are also presented in Table 7-2 and Table 7-3.

The limit load analyses consider material properties (yield and ultimate strength) of the base metal, and not the material properties of the weld metal. The base metal (piping) is considered to have more limiting material properties than the weld metal. Therefore, in the limit load evaluation the faulted loads (include both the axial loads (including pressure) and the moment loads) from Table 3-2 were increased by the Z-correction factors to account for reduction of the material toughness due to the welding process used during construction consistently with the methodology of SRP 3.6.3. It is confirmed that the limit load analysis in this report bounds both the weld metal and base metal since the more limiting material properties of the base metals were used in combination with additional penalty Z-correction factor for the stainless-steel weld.

The welding process implemented at locations 1 and 3 is a combination of Gas Tungsten Arc Welding (GTAW) and Shielded Metal Arc Welding (SMAW). Location 1 of loop A envelopes the two types of welding process: GTAW between the RPVON and the spool piece, and SMAW at the other end of the spool piece. A Z-correction factor is not applicable for the GTAW welding process; therefore, the SMAW process governs these evaluations at locations 1 and 3. The welding process implemented at location 2 is Submerged Arc Welding (SAW). For LBB evaluations, SAW is more limiting for crack stability analysis compared to SMAW process. Therefore, a conservative approach is taken assuming a SAW process at location 4 for all loops. The Z-correction factor for the SMAW and SAW welding processes (References 7-3 and 7-5) are as follows:

Locations 1 and 3: $Z = 1.15 [1.0 + 0.013 (OD-4)]$ for SMAW
Locations 2 and 4: $Z = 1.30 [1.0 + 0.010 (OD-4)]$ for SAW

Where OD is the outer diameter of the pipe in inches.

The Z-correction factors were calculated for the critical locations using the dimensions given in Table 3-1. The Z-correction factors are 1.60 for location 1, 1.69 for location 2, 1.63 for location 3 and 1.72 for location 4.

In the J-integral evaluation, J_{app} was calculated based on the faulted loads in Table 3-2 without any Z-correction factors to account for reduction in fracture toughness. This is because the calculation of J_{Ic}, as part of the J-integral evaluations, already considers reduction in fracture toughness due to thermal aging of the CASS materials at normal operating temperature over extended operating periods. This reduction in fracture toughness is based on correlations in NUREG/CR-4513 Revisions 1 and 2 (References 7-6 and 7-7), which have determined lower bound fracture toughness as discussed in Section 4.3. Therefore, no additional Z-factors are necessary because the reduction in fracture toughness is already captured with the consideration of end-of-life (saturated) fracture toughness values from NUREG/CR-4513.

Therefore, the limit load analysis for CASS materials in this report considered the reduced fracture toughness of the weld (Z-correction factor), and the J-applied analysis considered the reduced fracture toughness of the thermally aged CASS material per References 7-6 and 7-7.

7.4 SG AND RPV NOZZLE ALLOY 82/182 WELDS

Alloy 82/182 or Alloy 82 welds which are susceptible to PWSCC are present at the RPVINs, SGIN's and SGON's for all loops, and at the RPVONs for loops B and C. As discussed in Section 2.1, for RPVON's, SGIN's and SGON's the potential PWSCC have been mitigated. The limit load evaluation for the unmitigated and mitigated weld locations (locations 1 for loops B and C; locations 3, 4 and 12 for all loops) is performed.

The typical material properties of the Alloy 82/182 DM weld material from Table 4-4 were considered for the limit load analysis at locations 1 for loops B and C; at locations 3, 4 and 12 for all loops. The limit load analysis of the Alloy 82/182 welds considered a crack morphology factor (Z-multiplication factor).

[

]a,c,e

The Z-multiplication factor of 1.21 for the Alloy 82/182 material was calculated at location 1, loops B and C; locations 3, 4 and 12 for all loops. Note that in the limit load calculation for these locations, the applicable Z-correction factors for SMAW and SAW were conservatively used instead of both, the Z-correction factor of 1.0 for GTAW welding process at Alloy 82 material locations and the additional Z-multiplication factor of 1.21 for Alloy 82 material. The Z- multiplication factor for Inconel 152 is 1.0.

As discussed in Section 6.4, an increased factor of 1.69 to account for the PWSCC as applicable is applied to the leakage flaw size calculation.

Table 7-4 provides summary results for Alloy 82/182 DM weld material including associated leakage flaw sizes from Table 6-3. Table 7-4 also shows the critical flaw size for the Inconel 152 weld for location 1 of the loop A hot leg.

7.5 REFERENCES

7-1 Kanninen, M. F., et al., "Mechanical Fracture Predictions for Sensitized Stainless Steel Piping with Circumferential Cracks," EPRI NP-192, September 1976.

7-2 Kumar, V., German, M. D. and Shih, C. P., "An Engineering Approach for Elastic-Plastic Fracture Analysis," EPRI Report NP-1931, Project 1237-1, Electric Power Research Institute, July 1981.

7-3 Standard Review Plan; Public Comment Solicited; 3.6.3 Leak-Before-Break Evaluation Procedures; Federal Register/Vol. 52, No. 167/Friday, August 28, 1987/Notices, pp. 32626-32633.

7-4 ASME Pressure Vessel and Piping Division Conference Paper PVP2008-61840, "Technical Basis for Revision to Section XI Appendix C for Alloy 600/82/182/132 Flaw Evaluation in Both PWR and BWR Environments," July 28-31, Chicago IL, USA.

7-5 NUREG-0800, Revision 1, Standard Review Plan: 3.6.3 Leak-Before-Break Evaluation Procedures, March 2007.

7-6 O. K. Chopra, "Estimation of Fracture Toughness of Cast Stainless Steels During Thermal Aging in LWR Systems," NUREG/CR-4513, Revision 1, U.S. Nuclear Regulatory Commission, Washington, DC, August 1994.

7-7 O. K. Chopra, "Estimation of Fracture Toughness of Cast Stainless Steels During Thermal Aging in LWR Systems," NUREG/CR-4513, Revision 2, U.S. Nuclear Regulatory Commission, Washington, DC, May 2016" including Errata, March 15, 2021.

Table 7-1
VCSNS Unit 1 Stability Results
Based on Elastic-Plastic J-Integral Evaluations
for SA 351-CF8A

a,c,e

Table 7-2
VCSNS Unit 1 Stability Results
Based on Limit Load for SA376-TP304N Material

a,c,e

Table 7-3
VCSNS Unit 1 Stability Results
Based on Limit Load for SA351-CF8A Materials

a,c,e

Table 7-4
VCSNS Unit 1 Stability Results
Based on Limit Load for Alloy 82/182 and Inconel 152 Materials

a,c,e

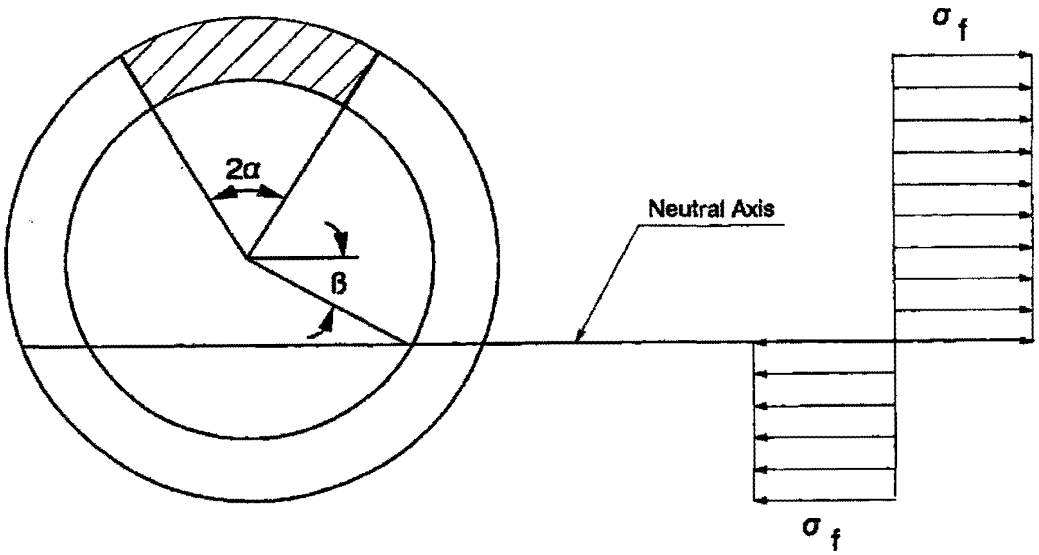


Figure 7-1

[ ]^{a,c,e} Stress Distribution

8.0 FATIGUE CRACK GROWTH ANALYSIS

To determine the sensitivity of the primary coolant system to the presence of small cracks, a plant specific fatigue crack growth (FCG) analysis was carried out for the [ ]^{a,c,e} region (see Location [ ]^{a,c,e} of Figure 3-2). This region was selected because crack growth calculated here will be typical (i.e., the design transient thermal and pressure stresses will be representative) of that in the entire primary loop. The crack growth at the [ ]^{a,c,e} will demonstrate that small surface flaws would not develop to through-wall flaws during the plant design life. Crack growths calculated at other locations can be expected to show less than 10% variation.

A [ ]^{a,c,e} of a plant typical in geometry and operational characteristics to any Westinghouse PWR System. [ ]^{a,c,e} All normal, upset, and test conditions were considered. A summary of the applied transients is provided in Table 8-1. Circumferentially oriented surface flaws were postulated in the region, assuming the flaw was located in three different locations, as shown in Figure 8-1. Specifically, these were:

Cross Section A: [ ]^{a,c,e}

Cross Section B: [ ]^{a,c,e}

Cross Section C: [ ]^{a,c,e}

Fatigue crack growth rate laws were used [ ]^{a,c,e}

The law for stainless steel was derived from Reference 8-1, with a very conservative correction for the R ratio, which is the ratio of minimum to maximum stress during a transient. For stainless steel, the fatigue crack growth formula is:

$$\frac{da}{dn} = \left(5.4\text{x}10^{-12}\right) \cdot K_{eff}^{\ 4.48} \text{ inches/cycle} \qquad (8\text{-}1)$$

Where:

da/dn = crack growth rate

K_{eff} = $K_{Imax} \times (1.0 - R)^{0.5}$

R = ratio of minimum K_I and maximum K_I

= K_{Imin}/K_{Imax}

[

]a,c,e

[ ]a,c,e (8-2)

Where:

[ ]a,c,e

Fatigue crack growth results for the Inconel 182 and Inconel 152 welds are expected to be about the same as Inconel 600 weld.

The calculated fatigue crack growth for semi-elliptic surface flaws of circumferential orientation and various depths is summarized in Table 8-2, and shows that the crack growth is very small, [ ].a,c,e To demonstrate that the small surface flaws will not result in a through-wall flaw over the design life of the plant. The aspect ratio for the postulated initial crack sizes are for a typical flaw shape of [ ]a,c,e (flaw length/flaw depth). Various initial flaw depths were considered in the FCG analysis to demonstrate that small, NDE-detectable flaw sizes on the order of [ ]a,c,e would be acceptable for the life of the plant (i.e., will not grow to the become complete through-wall).

The intent of FCG in the LBB analysis was not to use initial flaw depths that are larger than the Acceptance Tables of ASME Section XI IWB-3410-1, but rather to show a defense in-depth fatigue crack growth based on small flaw sizes that are detectable based on NDE examination techniques, which would not become through-wall flaws over the design life of the plant.

It should be noted that an underlying main assumption of this FCG analysis is that the design transients and associated occurrences which had been established for a 40-year design life remain applicable for the 60-year license renewal period. As expected, increase of the cycles for certain transients used in the FCG evaluation from 60-year to 80-year subsequent license renewal period may occur. However, the potential of exceeding the allowable transients for 40-year design basis is minimal. Furthermore, if some exceedances of the cycles for the 80-year design transients occur, the fatigue crack growth results documented in Table 8-2 show that there is a sufficient margin to ensure that small surface flaws will not become through-wall flaws. Additionally, the fatigue crack growth evaluation is considered a defense in depth review. FCG is no longer a requirement for the Leak-Before-Break (LBB) analysis, since the LBB analysis is based on the postulation of through-wall flaw, whereas the FCG analysis is performed based on the surface flaw. Furthermore, Reference 8-4 has indicated that "the Commission deleted the fatigue crack growth analysis in the proposed rule. This requirement was found to be unnecessary because it was bounded by the crack stability analysis." Nevertheless, the fatigue crack growth analysis is retained herein for information purposes and to demonstrate that small surface flaws do not result in through-wall flaws over the life of the plant.

8.1 REFERENCES

8-1 Bamford, W. H., "Fatigue Crack Growth of Stainless Steel Piping in a Pressurized Water Reactor Environment," Trans. ASME Journal of Pressure Vessel Technology, Vol. 101, Feb. 1979.

8-2 [ ][a,c,e]

8-3 [ ][a,c,e]

8-4 Nuclear Regulatory Commission, 10 CFR 50, Modification of General Design Criteria 4 Requirements for Protection Against Dynamic Effects of Postulated Pipe Ruptures, Final Rule, Federal Register/Vol. 52, No. 207/Tuesday, October 27, 1987/Rules and Regulations, pp. 41288-41295.

Table 8-1
Summary of Reactor Vessel Transients for VCSNS Unit 1

Number	Typical Transient Identification	Number of Cycles
	Normal Conditions	
1	Heatup and Cooldown at 100°F/hr. (pressurizer cooldown 200°F/hr.)	200
2	Load Follow Cycles (Unit loading and unloading at 5% of full power/min.)	18300
3	Step Load Increase and Decrease	2000
4	Large Step Load Decrease, with Steam Dump	200
5	Steady State Fluctuations	1000000
	Upset Conditions	
6	Loss of Load, without Immediate Turbine or Reactor Trip	80
7	Loss of Power (blackout with natural circulation in the Reactor Coolant System)	40
8	Loss of Flow (partial loss of flow, one pump only)	80
9	Reactor Trip from Full Power	400
	Test Conditions	
10	Turbine Roll Test	10
11	Hydrostatic Test Conditions	
	Primary Side	5
	Primary Side Leak Test	50
12	Cold Hydrostatic Test	10

Table 8-2 Typical Fatigue Crack Growth at [ ]a,c,e (40, 60 and 80-Years)			
Initial Flaw (in.)	**Final Flaw (in.)**		
	[ ]a,c,e	[ ]a,c,e	[ ]a,c,e
0.292	0.31097	0.30107	0.30698
0.300	0.31949	0.30953	0.31626
0.375	0.39940	0.38948	0.40763
0.425	0.45271	0.44350	0.47421

a,c,e

Figure 8-1
Typical Cross-Section of [ ]a,c,e

a,c,e

[ ]

Figure 8-2
Reference Fatigue Crack Growth Curves for [ ]a,c,e

a,c,e

Figure 8-3
Reference Fatigue Crack Growth Law for [ ]^{a,c,e} in a Water Environment at 600°F

9.0 ASSESSMENT OF MARGINS

The results of the leak rates of Section 6.4 and the corresponding stability and fracture toughness evaluations of Sections 7.1, 7.2, 7.3 and 7.4 are used in performing the assessment of margins. Margins are shown for piping SA376-TP304N material in Table 9-1, for elbow SA351-CF8A material in Table 9-2 and for Alloy 82/182 and Inconel 152 weld material in Table 9-3. All of the LBB recommended margins are satisfied. Also, the existence of Alloy 82 DM welds at the SGIN and SGON to safe-end, and Alloy 82/182 DM welds at the RPVIN to safe-end for loops A, B and C, and RPVON to safe-end locations for loops B and C are acceptable for the SLR program for 80-years of plant operation.

In summary, at all the critical locations relative to:

1. <u>Flaw Size</u> - Using faulted loads obtained by the absolute sum method, a margin of 2 or more exists between the critical flaw and the flaw having a leak rate of 10 gpm (the leakage flaw).

2. <u>Leak Rate</u> - A margin of 10 exists between the calculated leak rate from the leakage flaw and the plant leak detection capability of 1 gpm.

3. <u>Loads</u> - At the critical locations the leakage flaw was shown to be stable using the faulted loads obtained by the absolute sum method (i.e., a flaw twice the leakage flaw size is shown to be stable; hence the leakage flaw size is stable). A margin of 1 on loads using the absolute summation of faulted load combinations is satisfied.

Table 9-1
VCSNS Unit 1 Leakage Flaw Sizes, Critical Flaw Sizes, and Margins based on Limit Load Evaluation for SA376-TP304N Material

a,c,e

Table 9-2
VCSNS Unit 1 Leakage Flaw Sizes, Critical Flaw Sizes, and Margins based on Limit Load and J-Integral Evaluation for SA351-CF8A Material

a,c,e

Table 9-3
VCSNS Unit 1 Leakage Flaw Sizes, Critical Flaw Sizes, and Margins based on Limit Load Evaluation for Alloy 82/182 and Inconel 152 Materials

a,c,e

10.0 CONCLUSIONS

This report justifies the elimination of RCS primary loop pipe breaks from the structural design basis for the VCSNS Unit 1 for the 80-year license renewal period (SLR) as follows:

a. Stress corrosion cracking is precluded by use of fracture resistant materials in the piping system and controls on reactor coolant chemistry, temperature, pressure, and flow during normal operation. Alloy 82/182 or Alloy 82 welds are present at the RPVINs, SGIN's and SGON's for all loops, and at the RPVONs for loops B and C. The Alloy 82/182 welds are susceptible to PWSCC (Primary Water Stress Corrosion Cracking). The Alloy 82/182 weld at the RPVON for loop A is replaced with Inconel 152 weld, therefore no further consideration of the PWSCC effects is required for the SLR program.

b. To mitigate PWSCC effect due to the existence of Alloy 82/182, MSIP is applied at the RPVON's locations for loop B and C; the Alloy 82/182 DM weld at the RPVON to safe-end locations for loop A is replaced with Inconel 152 weld; and Alloy 152 inlay are installed on the inside surface of the Alloy 82 DW at the SG nozzle to safe-end locations.

   The LBB has been reevaluated for 80-year plant life SLR program at the mitigated DM weld locations, including the RPVON for loops B and C with applications of MSIP and the SGIN's and SGON's with Alloy 152 inlay. The LBB evaluation has been performed considering Alloy 82/182 material properties which includes appropriate PWSCC crack morphology parameter.

c. Evaluation of the RCS piping considering the thermal aging effects for the 80-year plant life period of the SLR program, and also the use of the most limiting fracture toughness properties ensures that each materials profile is appropriately bounded by the LBB results presented in this report. As stated in Section 7.0, for local and global failure mechanisms, all locations are evaluated using the cast stainless steel material properties (SA351-CF8A) which present a limiting condition due to the thermal aging effects.

d. Water hammer should not occur in the RCS piping because of system design, testing, and operational considerations.

e. The effects of low and high cycle fatigue on the integrity of the primary piping are negligible.

f. Adequate margin exists between the leak rate of small stable flaws and the capability of the VCSNS Unit 1 reactor coolant system pressure boundary Leakage Detection System.

g. Ample margin exists between the small stable flaw sizes of item (f) and larger stable flaws.

h. Ample margin exists in the material properties used to demonstrate end-of-service life (fully aged) stability of the critical flaws.

For the critical locations, postulated flaws will be stable because of the ample margins described in f, g, and h above.

Based on the discussion above, the Leak-Before-Break conditions and margins are satisfied for the VCSNS primary loop piping. All the recommended margins are satisfied. It is therefore concluded that dynamic effects of RCS primary loop pipe breaks need not be considered in the structural design basis for VCSNS Unit 1 for the 80-year plant life (subsequent license renewal program).

APPENDIX A: LIMIT MOMENT

[

] a,c,e

a,c,e

Figure A-1
Pipe with a Through-Wall Crack in Bending

Serial No.: 23-193
Docket No.: 50-395

Enclosure 4
Attachment 3

WCAP-18728-NP, REVISION 5

Virgil C. Summer (VCSNS) Unit 1
Dominion Energy South Carolina, Inc. (DESC)

WCAP-18728-NP
Revision 5

June 2023

V.C. Summer Nuclear Station Unit 1 Subsequent License Renewal: Evaluation of Reactor Vessel Integrity Time-Limited Aging Analyses

Westinghouse

WCAP-18728-NP
Revision 5

V.C. Summer Nuclear Station Unit 1 Subsequent License Renewal: Evaluation of Reactor Vessel Integrity Time-Limited Aging Analyses

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RECORD OF REVISION

Revision	Description	Completed
0	Original Issue	May 2022
1	Incorporate Dominion's comments on Revision 0. Changes are marked with revision bars.	June 2022
2	Incorporate Dominion's comments on Revision 1. Changes are marked with revision bars.	July 2022
3	Incorporate Dominion's comments on CGE-RV000-TM-ME-000004 Revision 1 on the surveillance capsule schedule in Section 7. Changes are marked with revision bars.	August 2022
4	Incorporate a comment from Dominion on CGE-GENW-TR-LG-000002 Revision 0-B addressing the USE positions in Section 5. Changes are marked with revision bars.	May 2023
5	Incorporate a comment from Dominion to address low temperature overpressure protection (LTOP) in Table 1-1 and address the disposition of nozzle P-T Limit curves as requested by Dominion. Changes are marked with revision bars.	June 2023

TABLE OF CONTENTS

LIST OF TABLES iv

LIST OF FIGURES vi

EXECUTIVE SUMMARY vii

1 TIME-LIMITED AGING ANALYSIS 1-1

2 CALCULATED FLUENCE 2-1

3 MATERIAL PROPERTY INPUT 3-1

4 PRESSURIZED THERMAL SHOCK 4-1

5 UPPER-SHELF ENERGY 5-1

6 HEATUP AND COOLDOWN PRESSURE-TEMPERATURE LIMIT CURVES 6-1
6.1 ADJUSTED REFERENCE TEMPERATURES CALCULATION 6-1
6.2 P-T LIMIT CURVES APPLICABILITY 6-14

7 SURVEILLANCE CAPSULE WITHDRAWAL SCHEDULES 7-1

8 REFERENCES 8-1

APPENDIX A CREDIBILITY EVALUATION OF THE VCSNS UNIT 1 SURVEILLANCE PROGRAM A-1

APPENDIX B EMERGENCY RESPONSE GUIDELINES B-1

LIST OF TABLES

Table 1-1 Evaluation of Time-Limited Aging Analyses Per the Criteria of 10 CFR 54.3 1-1

Table 2-1 Calculated Fast Neutron (E > 1.0 MeV) Fluence at the Surveillance Capsule Center for VCSNS Unit 1 2-2

Table 2-2 VCSNS Unit 1 – Maximum Fast Neutron (E > 1.0 MeV) Fluence Experienced by the Pressure Vessel Materials in the Beltline and Extended Beltline Regions 2-4

Table 3-1 Best-Estimate Cu and Ni Weight Percent Values, Initial RT_{NDT} Values, and Initial USE Values for the VCSNS Unit 1 RPV Beltline and Extended Beltline Materials 3-4

Table 3-2 Calculation of Position 2.1 CF Values for VCSNS Unit 1 Surveillance Materials 3-5

Table 3-3 Summary of the VCSNS Unit 1 RPV Beltline, Extended Beltline, and Surveillance Material CF Values based on Regulatory Guide 1.99, Revision 2, Position 1.1 and Position 2.1 3-6

Table 4-1 RT_{PTS} Calculations for VCSNS Unit 1 at 72 EFPY 4-2

Table 5-1 Predicted USE Values at 72 EFPY for the VCSNS Unit 1 Beltline and Extended Beltline Materials 5-3

Table 6-1 VCSNS Unit 1 Fluence and Fluence Factor Values for the Surface, 1/4T, and 3/4T Locations at 56 EFPY 6-2

Table 6-2 VCSNS Unit 1 Fluence and Fluence Factor Values for the Surface, 1/4T, and 3/4T Locations at 72 EFPY 6-3

Table 6-3 Calculation of the VCSNS Unit 1 ART Values at the 1/4T Location for the Reactor Vessel Beltline and Extended Beltline Materials at the End of PEO (56 EFPY) 6-4

Table 6-4 Calculation of the VCSNS Unit 1 ART Values at the 3/4T Location for the Reactor Vessel Beltline and Extended Beltline Materials at the End of PEO (56 EFPY) 6-6

Table 6-5 Calculation of the VCSNS Unit 1 ART Values for the Reactor Vessel Extended Beltline Nozzle Materials at the End of PEO (56 EFPY) 6-8

Table 6-6 Calculation of the VCSNS Unit 1 ART Values at the 1/4T Location for the Reactor Vessel Beltline and Extended Beltline Materials at the End of SPEO (72 EFPY) 6-9

Table 6-7 Calculation of the VCSNS Unit 1 ART Values at the 3/4T Location for the Reactor Vessel Beltline and Extended Beltline Materials at the End of SPEO (72 EFPY) 6-11

Table 6-8 Calculation of the VCSNS Unit 1 ART Values for the Reactor Vessel Extended Beltline Nozzle Materials at the End of SPEO (72 EFPY) 6-13

Table 6-9 Summary of the Limiting ART Values 6-14

Table 7-1 VCSNS Unit 1 Recommended Surveillance Capsule Withdrawal Schedule 7-3

Table A-1 Regulatory Guide 1.99, Revision 2, Credibility Criteria A-1

Table A-2 Calculation of Interim Chemistry Factors for the Credibility Evaluation Using VCSNS Unit 1 Surveillance Capsule Data Only A-4

Table A-3 VCSNS Unit 1 Surveillance Capsule Data Scatter about the Best-Fit Line A-5

Table B-1 Evaluation of VCSNS Unit 1 ERG Limit Category B-1

LIST OF FIGURES

Figure 2-1 Axial Boundary of the 1.0E+17 n/cm^2 Fast Neutron (E > 1.0 MeV) Fluence Threshold in the +Z Direction at 33.75 (end of Cycle 27), 54, and 72 EFPY 2-6

Figure 3-1 RPV Schematic for VCSNS Unit 1 3-3

Figure 5-1 Regulatory Guide 1.99, Revision 2, Position 1.2 & 2.2 Predicted Decrease in Upper-Shelf Energy as a Function of Copper and Fluence for VCSNS Unit 1 at theEnd of SPEO (72 EFPY) 5-5

Figure 7-1 Original Arrangement of Surveillance Capsules in the VCSNS Unit 1 Reactor Vessel .. 7-4

EXECUTIVE SUMMARY

This report presents the reactor vessel integrity Time-Limited Aging Analyses (TLAA) evaluations for the Virgil C. Summer Nuclear Station (VCSNS) Unit 1 reactor pressure vessel (RPV) in accordance with the requirements of the License Renewal Rule, 10 CFR Part 54. TLAAs are calculations that address safety-related aspects of the RPV within the bounds of the current 60-year license. These calculations must also be evaluated to account for an extended period of operation (80 years) also termed Subsequent Period of Extended Operation (SPEO).

VCSNS Unit 1 is currently licensed through 60 years of operation; therefore, with a 20-year license extension, the SPEO is applicable through 80 years of operation. The 60-year TLAAs evaluated in this report are applicable through 56 Effective Full Power Years (EFPY), which is deemed end of the Period of Extended Operation (PEO). Similarly, evaluations in this report performed at 80 years of operation are applicable through 72 EFPY (90% capacity factor of 80 years), which is deemed the end of SPEO. Updated neutron fluence evaluations were performed and documented in WCAP-18709-NP (Reference 5), as well as in Section 2 of this report. Updated neutron fluence evaluations were used to identify the VCSNS Unit 1 extended beltline materials and as input to the reactor vessel (RV) integrity evaluations in support of current plant operations and subsequent license renewal.

In addition to the RV integrity TLAA evaluations, the VCSNS Unit 1 surveillance data credibility evaluation is contained in Appendix A of this report. While not a TLAA, Appendix B provides an Emergency Response Guidelines (ERG) assessment for VCSNS Unit 1 for completeness.

A summary of results for the VCSNS Unit 1 TLAA evaluation is provided below. Based on the results of this TLAA evaluation, it is concluded that the VCSNS Unit 1 RV will continue to meet regulatory requirements through the SPEO.

Fluence

The RV beltline and extended beltline neutron fluence values applicable to a postulated 20-year license renewal period were calculated for the VCSNS Unit 1 materials. All transport calculations were carried out using the three-dimensional discrete ordinates code RAPTOR-M3G and the BUGLE-96 cross-section library. The analysis methodologies follow the guidance in Regulatory Guide 1.190 (Reference 2). It is also consistent with the methodology described in WCAP-18124-NP-A (Reference 4) that was generically approved by the United States Nuclear Regulatory Commission (USNRC) for calculating exposures of the RPV beltline (i.e., in general, RPV materials opposite the active fuel). See Section 2 for more details.

Pressurized Thermal Shock

All of the beltline and extended beltline materials in the VCSNS Unit 1 RV are projected to remain below the RT_{PTS} screening criteria values of 270°F for base metal and/or longitudinal welds and 300°F for circumferentially oriented welds (per 10 CFR 50.61) through SPEO (72 EFPY). See Section 4 for more details.

Upper-Shelf Energy

All of the beltline and extended beltline materials in the VCSNS Unit 1 RV are projected to remain above the upper-shelf energy (USE) screening criterion value of 50 ft-lb (per 10 CFR 50, Appendix G), through SPEO (72 EFPY). See Section 5 for more details.

Adjusted Reference Temperatures and P-T Limit Curves Applicability Check

Adjusted Reference Temperatures (ARTs) are calculated for the end of PEO at 56 EFPY and for the end of SPEO at 72 EFPY in order to perform an applicability check on the existing pressure-temperature (P-T) limit curves for VCSNS Unit 1. With the consideration of TLAA fluence projections, revised Position 2.1 chemistry factor values, and recalculated initial RT_{NDT} values, the applicability of the VCSNS Unit 1 cylindrical shell P-T limit curves currently in the Technical Specifications remain applicable through 72 EFPY. The conclusion considers the RV inlet/outlet nozzles.

Surveillance Capsule Withdrawal Schedule

With consideration of a 20-year license renewal to 80 years of operation (72 EFPY), Capsule Y, which currently resides in the spent fuel pool, must be reinserted for additional irradiation. The surveillance capsule withdrawal schedule in Table 7-1 identifies the additional exposure required by the capsule in order to meet the guidance of NUREG-2191 (Reference 18) (GALL-SLR) for a capsule to be withdrawn and tested between one and two times the peak RV wall neutron fluence at the end of SPEO. See Section 7 for more details.

1 TIME-LIMITED AGING ANALYSIS

Time-Limited Aging Analyses (TLAAs) are those licensee calculations that:

1. Involve systems, structures, and components (SSCs) within the scope of license renewal
2. Consider the effects of aging.
3. Involve time-limited assumptions defined by the current operating term (e.g., 60 years).
4. Were determined to be relevant by the licensee in making a safety determination.
5. Involve conclusions or provide the basis for conclusions related to the capability of the SSC to perform its intended functions.
6. Are contained or incorporated by reference in the current licensing basis (CLB).

The potential TLAAs for the reactor pressure vessel (RPV) are identified in Table 1-1 along with indication of whether or not they meet the six (6) criteria of 10 CFR 54.3 (Reference 1) for TLAAs.

Table 1-1 Evaluation of Time-Limited Aging Analyses Per the Criteria of 10 CFR 54.3

Time-Limited Aging Analysis	Calculated Fluence	Pressurized Thermal Shock[(a)]	Upper-Shelf Energy	Pressure-Temperature Limits for Heatup and Cooldown	Low Temperature Overpressure Protection
Involves SSC Within the Scope of License Renewal	YES	YES	YES	YES	YES
Considers the Effects of Aging	YES	YES	YES	YES	YES
Involves Time-Limited Assumptions Defined by the Current Operating Term	YES	YES	YES	YES	YES
Determined to be Relevant by the Licensee in Making a Safety Determination	YES	YES	YES	YES	YES
Involves Conclusions or Provides the Basis for Conclusions Related to the Capability of SSC to Perform Its Intended Function	YES	YES	YES	YES	YES
Contained or Incorporated by Reference in the CLB	YES	YES	YES	YES	YES

Note:

(a) The limiting Pressurized Thermal Shock (PTS) values are used to determine the appropriate Emergency Response Guideline (ERG) Limits category for VCSNS Unit 1 through the end of the potential subsequent license extension period. However, ERG limits are outside the scope of 10 CFR Part 54.3. ERG limits are discussed in Appendix B.

2 CALCULATED FLUENCE

Estimated RPV beltline and extended beltline fast neutron (E > 1.0 MeV) fluences at the end of 80 years of operation were calculated for VCSNS Unit 1 in WCAP-18709-NP. The analyses methodologies used to calculate the VCSNS Unit 1 RPV fluences followed the guidance of Regulatory Guide 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence" (Reference 2). These methodologies have been approved by the USNRC for the beltline region, i.e., materials directly surrounding the core and adjacent materials per 10 CFR 50, Appendix G (Reference 3), which are projected to experience the highest fluence. The methodologies, along with the NRC safety evaluation, are contained in detail in WCAP-18124-NP-A (Reference 4). For VCSNS Unit 1, the beltline region has traditionally included the intermediate and lower shell forgings, and the circumferential welds between these components. The traditional beltline and extended beltline materials are identified in Table 2-2 and Figure 2-1.

Materials exceeding a fast neutron (E > 1.0 MeV) fluence of 1.0×10^{17} n/cm^2 at the end of the SPEO are evaluated for changes in fracture toughness. RPV materials that are not traditionally plant-limiting because of low levels of neutron radiation must now be evaluated to determine the accumulated fluence at the end of SPEO. Therefore, fast neutron (E > 1.0 MeV) fluence calculations were performed for the VCSNS Unit 1 RPV to determine where it will exceed a fast neutron (E > 1.0 MeV) fluence of 1.0×10^{17} n/cm^2 at the end of SPEO. The materials that exceed the 1.0×10^{17} n/cm^2 fast neutron (E > 1.0 MeV) fluence threshold and were not evaluated in past analyses of record as part of the traditional beltline, are referred to as extended beltline materials in this report and are evaluated to determine the effect of neutron irradiation embrittlement during SPEO.

All the transport calculations were carried out using the three-dimensional discrete ordinates code RAPTOR-M3G and the BUGLE-96 cross-section library. The BUGLE-96 library provides a 67-group coupled neutron-gamma ray cross-section data set produced specifically for light water reactor applications. In these analyses, anisotropic scattering was treated with a P_3 Legendre expansion and the angular discretization was modeled with an S_{16} order of angular quadrature. Energy- and space-dependent core power distributions, as well as system operating temperatures, were treated on a fuel-cycle-specific basis.

The calculations for fuel Cycles 1 through 26 determine the neutron exposure of the pressure vessel and surveillance capsules based on completed fuel cycles. The projection for Cycle 27 is based on the actual loading, but yet to be completed, fuel cycle. The projections for Cycle 28 and beyond, up to and including the end of PEO (56 EFPY) and the end of SPEO (72 EFPY), are based on the average core power distributions and reactor operating conditions of Cycles 25, 26, and 27 and are determined both with and without a 10% positive bias on the peripheral and re-entrant corner assembly relative powers.

Table 2-1 gives the VCSNS Unit 1 calculated fast neutron (E > 1.0 MeV) fluences at the capsule locations including all withdrawn surveillance capsules (Capsules U, V, X, W, Y, and Z).

Table 2-2 presents the fast neutron (E > 1.0 MeV) fluence results for the applicable portions of the pressure vessel from the neutron transport analyses. From Table 2-2 it is observed that outlet nozzles and inlet nozzles have fast neutron (E > 1.0 MeV) fluence greater than 1.0×10^{17} n/cm^2 at the lowest extent of the nozzle forging to nozzle shell weld at 72 EFPY. All materials located above the nozzles will remain below 1.0×10^{17} n/cm^2 through 72 EFPY. Table 2-2 indicates that the lower shell to lower vessel head circumferential weld, and all materials below it, will remain below 1.0×10^{17} n/cm^2 through SPEO. Figure 2-1 shows the axial boundary of the 1.0×10^{17} n/cm^2 fluence threshold (at 54 EFPY and 72 EFPY) as a function of azimuthal position (Z versus θ).

All data presented in this section, along with additional details, are presented in WCAP-18709-NP (Reference 5). This includes description of uncertainties and validation of the analytical model based on the measured plant dosimetry.

Table 2-1 Calculated Fast Neutron (E > 1.0 MeV) Fluence at the Surveillance Capsule Center for VCSNS Unit 1[(a)]

Cycle	Cycle Length (EFPY)	Cumulative Operating Time (EFPY)	Fluence (n/cm^2)	
			17°	20°
1	1.13	1.13	6.75E+18[(b)]	5.90E+18
2	0.67	1.80	1.01E+19	8.94E+18
3	1.13	2.93	1.54E+19[(c)]	1.36E+19
4	1.16	4.09	2.03E+19	1.80E+19
5	0.95	5.04	2.51E+19[(d)]	2.24E+19
6	1.17	6.21	3.13E+19	2.79E+19
7	1.22	7.43	3.63E+19	3.24E+19
8	1.19	8.61	4.13E+19	3.69E+19
9	1.27	9.89	4.66E+19	4.15E+19
10	1.32	11.21	5.18E+19	4.63E+19[(e)]
11	1.36	12.56	5.78E+19	5.15E+19
12	1.37	13.94	6.35E+19	5.66E+19
13	1.09	15.03	6.76E+19	6.02E+19
14	1.33	16.36	7.34E+19	6.53E+19[(f)]
15	1.35	17.71	7.88E+19	7.01E+19[(g)]
16	1.34	19.05	8.42E+19	7.49E+19
17	1.38	20.43	9.00E+19	8.01E+19
18	1.30	21.73	9.52E+19	8.49E+19
19	1.31	23.04	1.01E+20	8.97E+19
20	1.36	24.41	1.06E+20	9.46E+19
21	1.29	25.70	1.11E+20	9.93E+19
22	1.29	26.99	1.17E+20	1.04E+20
23	1.34	28.33	1.23E+20	1.09E+20
24	1.32	29.65	1.28E+20	1.14E+20
25	1.36	31.01	1.34E+20	1.19E+20
26	1.37	32.38	1.39E+20	1.24E+20
27[(h)]	1.37	33.75	1.45E+20	1.30E+20
No bias on the peripheral and re-entrant corner assembly relative powers				
Future[(i)]	--	36.00	1.55E+20	1.38E+20
Future[(i)]	--	42.00	1.80E+20	1.61E+20
Future[(i)]	--	48.00	2.05E+20	1.83E+20

Table 2-1 Calculated Fast Neutron (E > 1.0 MeV) Fluence at the Surveillance Capsule Center for VCSNS Unit 1[(a)]

Cycle	Cycle Length (EFPY)	Cumulative Operating Time (EFPY)	Fluence (n/cm²)	
			17°	20°
Future[(i)]	--	54.00	2.30E+20	2.06E+20
Future[(i)]	--	60.00	2.55E+20	2.28E+20
Future[(i)]	--	66.00	2.80E+20	2.51E+20
Future[(i)]	--	72.00	3.05E+20	2.73E+20
+10% bias on the peripheral and re-entrant corner assembly relative				
Future[(i)]	--	36.00	1.56E+20	1.39E+20
Future[(i)]	--	42.00	1.83E+20	1.64E+20
Future[(i)]	--	48.00	2.11E+20	1.88E+20
Future[(i)]	--	54.00	2.38E+20	2.13E+20
Future[(i)]	--	60.00	2.66E+20	2.38E+20
Future[(i)]	--	66.00	2.93E+20	2.63E+20
Future[(i)]	--	72.00	3.21E+20	2.87E+20

Notes:

(a) Information taken from WCAP-18709-NP (Reference 5).

(b) This value is applicable to Capsule U.

(c) This value is applicable to Capsule V

(d) This value is applicable to Capsule X.

(e) This value is applicable to Capsule W.

(f) This value is applicable to Capsule Z.

(g) This value is applicable to Capsule Y.

(h) Cycle 27 was the current operating cycle at the time this summary report was authored.

(i) Values beyond Cycle 27 are based on the average core power distributions and reactor operating conditions of Cycles 25, 26, and 27 and are determined both with and without a 1.1 bias on the peripheral and re-entrant corner assembly relative powers.

Table 2-2 VCSNS Unit 1 – Maximum Fast Neutron (E > 1.0 MeV) Fluence Experienced by the Pressure Vessel Materials in the Beltline and Extended Beltline Regions

Projections with no bias on the peripheral and re-entrant corner assembly relative powers									
Material Location	**Material**	**Fast Neutron (E > 1.0 MeV) Fluence (n/cm^2)**							
		33.75 EFPY[a]	**36 EFPY**	**42 EFPY**	**48 EFPY**	**54 EFPY**	**60 EFPY**	**66 EFPY**	**72 EFPY**
Extended Beltline Materials	Inlet Nozzle to Nozzle Shell Weld (lowest extent)	1.41E+17	1.50E+17	1.75E+17	2.00E+17	2.24E+17	2.49E+17	2.74E+17	2.99E+17
	Inlet Nozzle Postulated 1/4T Flaw	1.68E+16	1.79E+16	2.09E+16	2.39E+16	2.68E+16	2.98E+16	3.28E+16	3.58E+16[d]
	Outlet Nozzle to Nozzle Shell Weld (lowest extent)	5.99E+16	6.39E+16	7.44E+16	8.49E+16	9.55E+16	1.06E+17	1.17E+17	1.27E+17
	Outlet Nozzle Postulated 1/4T Flaw	8.54E+15	9.11E+15	1.06E+16	1.21E+16	1.37E+16	1.52E+16	1.67E+16	1.82E+16[d]
	Nozzle Shell[b]	1.83E+18	1.95E+18	2.26E+18	2.58E+18	2.89E+18	3.20E+18	3.52E+18	3.83E+18
	Nozzle-to-Intermediate Shell Circumferential Weld	1.94E+18	2.07E+18	2.40E+18	2.74E+18	3.07E+18	3.40E+18	3.74E+18	4.07E+18
Beltline Materials	Intermediate Shell	4.14E+19	4.40E+19	5.10E+19	5.81E+19	6.51E+19	7.21E+19	7.92E+19	8.62E+19
	Intermediate Shell Longitudinal Weld – 45°/225°	1.40E+19	1.49E+19	1.72E+19	1.96E+19	2.19E+19	2.42E+19	2.66E+19	2.89E+19
	Intermediate-to-Lower Shell Circumferential Weld	4.13E+19	4.40E+19	5.10E+19	5.80E+19	6.51E+19	7.21E+19	7.91E+19	8.62E+19
	Lower Shell	4.14E+19	4.40E+19	5.11E+19	5.81E+19	6.52E+19	7.23E+19	7.93E+19	8.64E+19
	Lower Shell Longitudinal Weld – 135°/315°	1.42E+19	1.51E+19	1.75E+19	1.98E+19	2.22E+19	2.46E+19	2.70E+19	2.93E+19
Outside of beltline region	Lower Shell to Bottom Head Circumferential Weld[c]	4.69E+15	5.00E+15	5.81E+15	6.62E+15	7.43E+15	8.24E+15	9.05E+15	9.86E+15

Table 2-2 VCSNS Unit 1 – Maximum Fast Neutron (E > 1.0 MeV) Fluence Experienced by the Pressure Vessel Materials in the Beltline and Extended Beltline Regions

Projections with a +10% bias on the peripheral and re-entrant corner assembly relative powers

Material Location	Material	Fast Neutron (E > 1.0 MeV) Fluence (n/cm²)							
		33.75 EFPY[a]	36 EFPY	42 EFPY	48 EFPY	54 EFPY	60 EFPY	66 EFPY	72 EFPY
Extended Beltline Materials	Inlet Nozzle to Nozzle Shell Weld (lowest extent)	1.41E+17	1.51E+17	1.77E+17	2.04E+17	2.30E+17	2.57E+17	2.83E+17	3.10E+17
	Inlet Nozzle Postulated 1/4T Flaw	1.68E+16	1.80E+16	2.11E+16	2.43E+16	2.74E+16	3.06E+16	3.38E+16	3.69E+16[d]
	Outlet Nozzle to Nozzle Shell Weld (lowest extent)	5.99E+16	6.42E+16	7.54E+16	8.66E+16	9.79E+16	1.09E+17	1.20E+17	1.32E+17
	Outlet Nozzle Postulated 1/4T Flaw	8.54E+15	9.14E+15	1.07E+16	1.24E+16	1.40E+16	1.56E+16	1.72E+16	1.88E+16[d]
	Nozzle Shell[b]	1.83E+18	1.96E+18	2.30E+18	2.64E+18	2.98E+18	3.32E+18	3.66E+18	4.00E+18
	Nozzle-to-Intermediate Shell Circumferential Weld	1.94E+18	2.08E+18	2.44E+18	2.80E+18	3.16E+18	3.52E+18	3.89E+18	4.25E+18
Beltline Materials	Intermediate Shell	4.14E+19	4.42E+19	5.19E+19	5.96E+19	6.73E+19	7.50E+19	8.27E+19	9.04E+19
	Intermediate Shell Longitudinal Weld – 45°/225°	1.40E+19	1.50E+19	1.75E+19	2.01E+19	2.26E+19	2.52E+19	2.78E+19	3.03E+19
	Intermediate-to-Lower Shell Circumferential Weld	4.13E+19	4.42E+19	5.19E+19	5.96E+19	6.73E+19	7.50E+19	8.27E+19	9.04E+19
	Lower Shell	4.14E+19	4.42E+19	5.20E+19	5.97E+19	6.74E+19	7.52E+19	8.29E+19	9.06E+19
	Lower Shell Longitudinal Weld – 135°/315°	1.42E+19	1.52E+19	1.78E+19	2.04E+19	2.30E+19	2.56E+19	2.82E+19	3.08E+19
Outside of beltline region	Lower Shell to Bottom Head Circumferential Weld[c]	4.69E+15	5.03E+15	5.91E+15	6.79E+15	7.67E+15	8.55E+15	9.44E+15	1.03E+16

Notes:

(a) Value listed is the projected EFPY at the end of Cycle 27.

(b) Exposure values for the nozzle shell longitudinal welds are bounded by the exposure values for the nozzle shell (a.k.a. upper shell).

(c) Maximum exposure values occur at the RPV outer radius.

(d) While the fluence at this location is less than 1E+17 n/cm^2, it is identified as extended beltline since portions of the nozzle exceed the criterion.

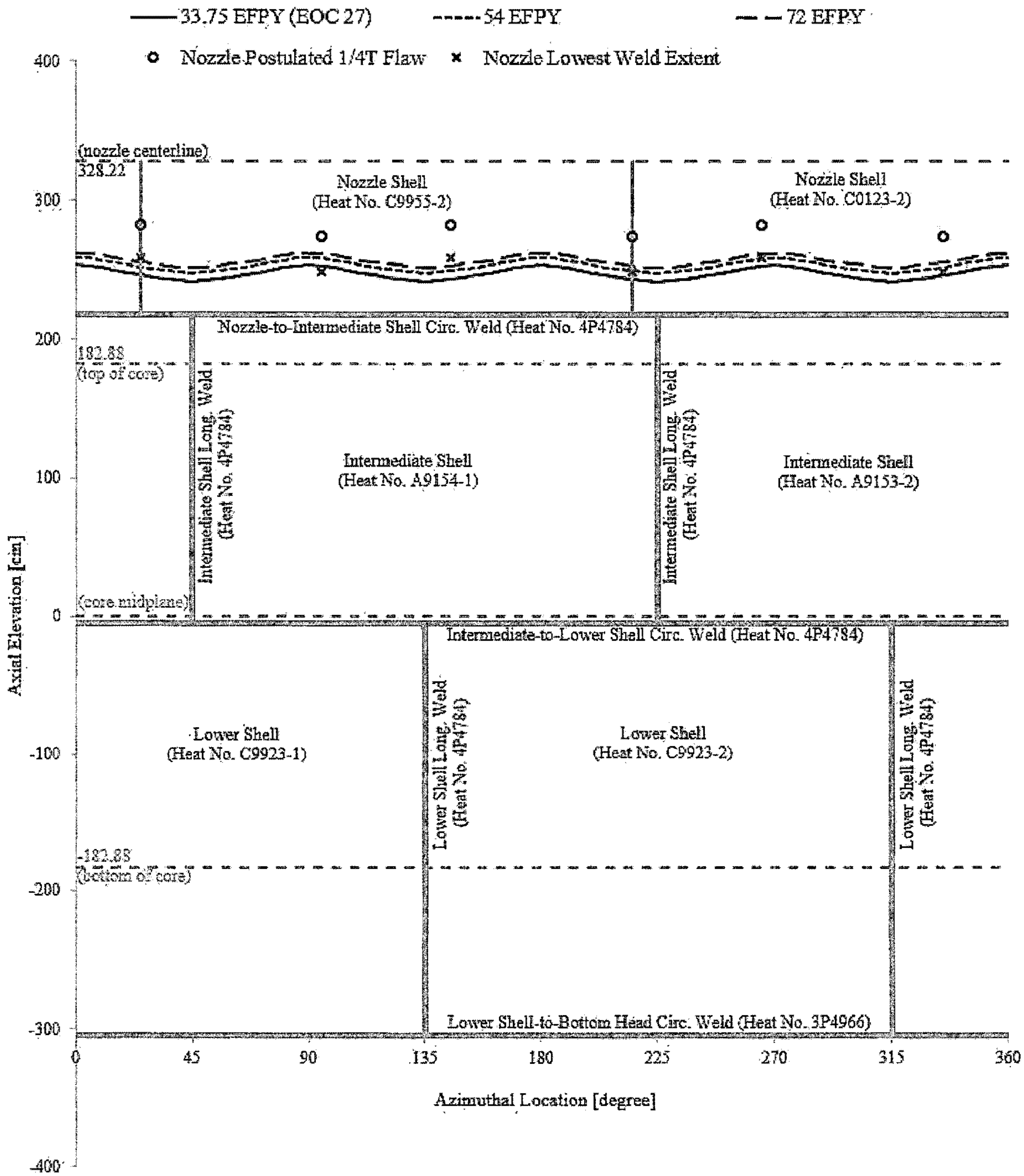

Figure 2-1 Axial Boundary of the 1.0E+17 n/cm² Fast Neutron (E > 1.0 MeV) Fluence Threshold in the +Z Direction at 33.75 (end of Cycle 27), 54, and 72 EFPY

3 MATERIAL PROPERTY INPUT

The requirements for RV integrity are specified in 10 CFR 50, Appendix G (Reference 3) and 10 CFR 50.61 (Reference 6). The beltline region of the RV is defined as the following in 10 CFR 50, Appendix G:

> *... the region of the reactor vessel (shell material including welds, heat affected zones, and plates or forgings) that directly surrounds the effective height of the active core and adjacent regions of the reactor vessel that are predicted to experience sufficient neutron radiation damage to be considered in the selection of the most limiting material with regard to radiation damage.*

The VCSNS Unit 1 beltline materials consist of two (2) Intermediate Shell Plates, two (2) Lower Shell Plates, and their associated welds. The VCSNS Unit 1 surveillance plate material was made from RV Intermediate Shell Plate 11-1, Heat # A9154-1. The VCSNS Unit 1 RV beltline welds were fabricated using weld wire Heat # 4P4784, Flux Type Linde 124, Flux Lot # 3930. The weld material in the VCSNS Unit 1 surveillance program was fabricated with the same material heat, flux type, and flux lot number.

Any RV materials that are predicted to experience a neutron fluence exposure greater than 1.0×10^{17} n/cm^2 (E > 1.0 MeV) at the end of the licensed operating period should be considered to experience neutron embrittlement. Based on the results of Section 2 of this report, the materials that exceeded the 1×10^{17} n/cm^2 (E > 1.0 MeV) threshold at 72 EFPY that were not included within the original beltline are considered to be the VCSNS Unit 1 extended beltline materials and are evaluated to determine their impact on the proposed SPEO of operation. The VCSNS Unit 1 RV extended beltline contains one (1) Nozzle Shell-to-Intermediate Shell circumferential weld, two (2) Nozzle Shell Plates (also termed upper shell), two (2) Nozzle Shell longitudinal welds, three (3) Inlet Nozzles, three (3) Outlet Nozzles, and the six (6) Nozzle-to-Nozzle Shell welds. Only those materials with a fluence greater than 1×10^{17} n/cm^2 (E > 1.0 MeV) at the end of SPEO require the effects of embrittlement to be included when evaluating the RV integrity.

The RV forgings/plates and weld materials are shown in Figure 3-1 for VCSNS Unit 1. Used in conjunction with the fluence data in Table 2-2, and Figure 2-1, the beltline and extended beltline materials are identified as shown in Table 3-1. Note that for RV welds, the terms "girth" and "circumferential" are used interchangeably; herein, these welds shall be referred to as circumferential welds. Similarly, for RV welds, the terms "axial" and "longitudinal" are used interchangeably; herein, these welds shall be referred to as longitudinal welds.

The unirradiated material property inputs used for the RV integrity evaluations herein are contained in PWROG-21037-NP (Reference 7). PWROG-21037-NP defined or redefined many of the material properties and chemistry values using the most up-to-date methodologies and all available data; therefore, the values utilized herein supersede previously documented values. The sources and methods used in the determination of the chemistry factors and the fracture toughness properties are summarized below.

Chemical Compositions

The best-estimate copper (Cu) and nickel (Ni) chemical compositions for the VCSNS Unit 1 beltline and extended beltline materials are presented in Table 3-1. The best-estimate weight percent copper and nickel values for the beltline and extended beltline materials were previously reported in PWROG-21037-NP and were included in RV integrity evaluations as part of this TLAA effort.

Fracture Toughness Properties

The most up-to-date initial RT_{NDT} and initial USE values are documented in PWROG-21037-NP for VCSNS Unit 1. The beltline and extended beltline material properties of the VCSNS Unit 1 RV are presented in Table 3-1 herein. The differences between the unirradiated RT_{NDT} values summarized in the FSAR and those determined herein are a result of a change in curve-fitting method (hand-drawn versus hyperbolic tangent) used to fit the Charpy V-notch test data.

Chemistry Factor Values

The chemistry factor (CF) values were calculated using Positions 1.1 and 2.1 of Regulatory Guide 1.99, Revision 2 (Reference 8). Position 1.1 uses Tables 1 and 2 from the Regulatory Guide along with the best-estimate copper and nickel weight percent values (contained in Table 3-1). Position 2.1 uses the surveillance capsule data from all capsules tested to date and surveillance data from other plants, as applicable. Credibility evaluations of the VCSNS Unit 1 surveillance data are provided in Appendix A of this report. The calculated capsule fluence values are provided in Table 2-1 and are used to determine the Position 2.1 CFs as shown in Table 3-2. Table 3-3 summarizes the Positions 1.1 and 2.1 CF values determined for the VCSNS Unit 1 RPV beltline and extended beltline materials.

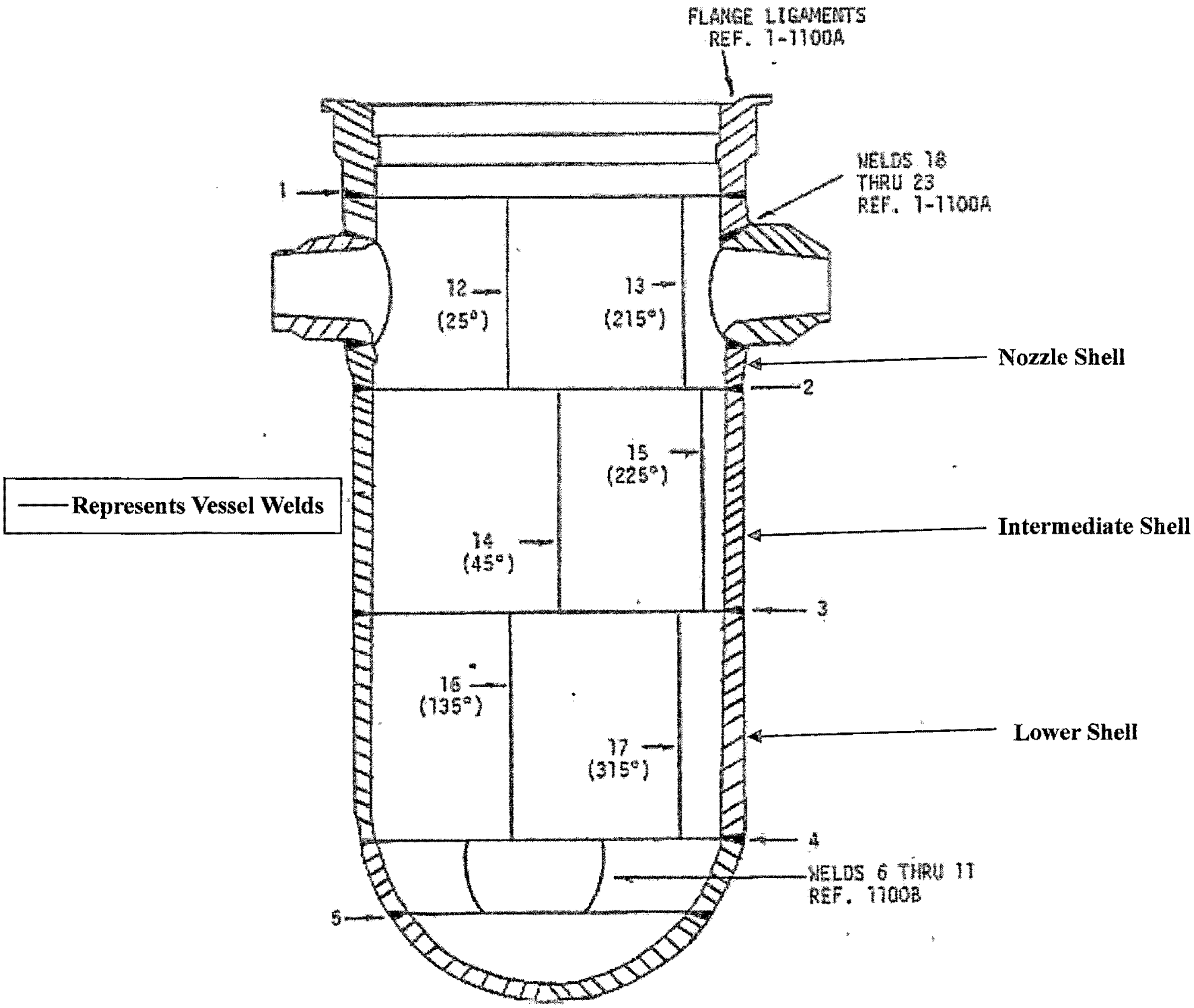

Figure 3-1 RPV Schematic for VCSNS Unit 1

Table 3-1 Best-Estimate Cu and Ni Weight Percent Values, Initial RT_{NDT} Values, and Initial USE Values for the VCSNS Unit 1 RPV Beltline and Extended Beltline Materials[(a)]

Material Identification	Wt. % Cu	Wt. % Ni	Initial RT_{NDT} (°F)	σ_I (°F)	Unirradiated USE (ft-lb)
Beltline					
Intermediate Shell 11-1 (Heat # A9154-1)	0.10	0.51	21	0	76
Intermediate Shell 11-2 (Heat # A9153-2)	0.09	0.45	-20	0	107
Lower Shell 10-1 (Heat # C9923-1)	0.08	0.41	5	0	106
Lower Shell 10-2 (Heat # C9923-2)	0.08	0.41	4	0	92
Intermediate Shell Long. Weld Seams BC & BD (Heat # 4P4784, Flux Type Linde 124, Lot # 3930)	0.05	0.91	-49	0	86
Intermediate to Lower Shell Circ. Weld Seam AB (Heat # 4P4784, Flux Type Linde 124, Lot # 3930)					
Lower Shell Long. Weld Seams BA & BB (Heat # 4P4784, Flux Type Linde 124, Lot # 3930)					
Extended Beltline					
Nozzle Shell 12-1 (Heat # C9955-2)	0.13	0.57	9	0	101
Nozzle Shell 12-2 (Heat # C0123-2)	0.12	0.58	15	0	91
Inlet Nozzle 436B-1 (Heat # Q2Q41W)	0.127[(b)]	0.76	-20	0	152
Inlet Nozzle 436B-2 (Heat # Q2Q39W)	0.127[(b)]	0.82	0	0	115
Inlet Nozzle 436B-3 (Heat # Q2Q39W)	0.127[(b)]	0.82	-20	0	138
Outlet Nozzle 437B-1 (Heat # Q2Q40)	0.127[(b)]	0.85	-10	0	159
Outlet Nozzle 437B-2 (Heat # Q2Q40W)	0.127[(b)]	0.80	-10	0	165
Outlet Nozzle 437B-3 (Heat # Q2Q44W)	0.127[(b)]	0.78	0	0	155
Nozzle to Intermediate Shell Circ. Weld Seam AC (Heat # 4P4784, Flux Type Linde 124, Lot # 3930)	0.05	0.91	-49	0	86
Nozzle Shell Long. Weld Seams BE and BF[(c)]	0.06[(c)]	1.01[(c)]	10[(c)]	0[(c)]	80[(c)]
Inlet/Outlet Nozzle Forgings to Nozzle Shell Weld Seams 15A/B/C & 16A/B/C[(c)]					
Surveillance Material[(d)]					
Intermediate Shell 11-1 (Heat # A9154-1)	-	-	-	-	-
Surveillance Weld (Heat # 4P4784, Flux Type Linde 124, Lot # 3930)	0.04	0.95	-	-	-

Notes:

(a) The information is extracted from PWROG-21037-NP (Reference 7). All values are based on information extracted from the V.C. Summer Unit 1 CMTRs and/or vessel fabrication records, unless noted otherwise.

(b) Generic value for SA-508 Class 2 nozzle forgings from PWROG-15109-NP-A (Reference 9).

(c) The specific heat number used in weld seams could not be identified. To address these situations, values were determine based on a review of all V.C. Summer weld heats used in the fabrication of the VCSNS Unit 1 RV. These generic values were defined in PWROG-21037-NP (Reference 7).

(d) Surveillance plate and weld data identified in WCAP-16298-NP (Reference 10).

Table 3-2 Calculation of Position 2.1 CF Values for VCSNS Unit 1 Surveillance Materials

Material	Capsule	Capsule Fluence[(a)] (x 10^{19} n/cm^2, E > 1.0 MeV)	FF[(b)]	Measured ΔRT_{NDT}[(c)] (°F)	FF*ΔRT_{NDT} (°F)	FF^2
Intermediate Shell 11-1 (Longitudinal)	U	0.675	0.890	36.1	32.1	0.792
	V	1.54	1.119	53.2	59.6	1.253
	X	2.51	1.247	38.3	47.8	1.555
	W	4.63	1.387	66.2	91.8	1.924
	Z	6.53	1.451	98.9	143.5	2.106
Intermediate Shell 11-1 (Transverse)	U	0.675	0.890	14.5	12.9	0.792
	V	1.54	1.119	32.1	35.9	1.253
	X	2.51	1.247	26.7	33.3	1.555
	W	4.63	1.387	57.8	80.2	1.924
	Z	6.53	1.451	87.0	126.3	2.106
				SUM:	663.4	15.261
	$CF_{11\text{-}1} = \Sigma(FF * \Delta RT_{NDT}) \div \Sigma(FF^2) = (663.4) \div (15.261) = 43.5°F$					
Surveillance Weld (Heat #4P4784)	U	0.675	0.890	28.6 (22.7)	25.4	0.792
	V	1.54	1.119	59.2 (47.0)	66.3	1.253
	X	2.51	1.247	28.6 (22.7)	35.7	1.555
	W	4.63	1.387	54.8 (43.5)	76.0	1.924
	Z	6.53	1.451	82.2 (65.2)	119.2	2.106
				SUM:	322.7	7.630
	$CF_{Surv.\ Weld} = \Sigma(FF * \Delta RT_{NDT}) \div \Sigma(FF^2) = (322.7) \div (7.630) = 42.3°F$					

Notes:

(a) Fluence taken from Table 2-1.

(b) FF = fluence factor = $f^{(0.28 - 0.10*\log(f))}$.

(c) Measured ΔRT_{NDT} taken from WCAP-16298-NP (Reference 10). The VCSNS Unit 1 surveillance weld measured ΔRT_{NDT} results have been adjusted by a ratio of 1.26 to account for chemistry differences between the Heat # 4P4784 surveillance weld (CF = 54°F) and RV welds (CF = 68°F). The unadjusted measured ΔRT_{NDT} values are listed in parentheses.

Table 3-3 Summary of the VCSNS Unit 1 RPV Beltline, Extended Beltline, and Surveillance Material CF Values based on Regulatory Guide 1.99, Revision 2, Position 1.1 and Position 2.1

Material Description	Chemistry Factor	
	Position 1.1[a] (°F)	Position 2.1[b] (°F)
Beltline		
Intermediate Shell 11-1 (Heat # A9154-1)	65.0	43.5
Intermediate Shell 11-2 (Heat # A9153-2)	58.0	-
Lower Shell 10-1 (Heat # C9923-1)	51.0	-
Lower Shell 10-2 (Heat # C9923-2)	51.0	-
Intermediate Shell Long. Weld Seams BC & BD (Heat # 4P4784)	68.0	42.3
Intermediate to Lower Shell Circ. Weld Seam AB (Heat # 4P4784)	68.0	42.3
Lower Shell Long. Weld Seams BA & BB (Heat # 4P4784)	68.0	42.3
Extended Beltline		
Nozzle Shell 12-1 (Heat # C9955-2)	90.1	-
Nozzle Shell 12-2 (Heat # C0123-2)	82.6	-
Inlet Nozzle 436B-1 (Heat # Q2Q41W)	92.1	-
Inlet Nozzle 436B-2 (Heat # Q2Q39W)	93.0	-
Inlet Nozzle 436B-3 (Heat # Q2Q39W)	93.0	-
Outlet Nozzle 437B-1 (Heat # Q2Q40)	93.0	-
Outlet Nozzle 437B-2 (Heat # Q2Q40W)	93.0	-
Outlet Nozzle 437B-3 (Heat # Q2Q44W)	92.6	-
Nozzle to Intermediate Shell Circ. Weld Seam AC (Heat # 4P4784)	68.0	42.3
Nozzle Shell Long. Weld Seams BE and BF	82.0	-
Inlet/Outlet Nozzle Forgings to Nozzle Shell Weld Seams 15A/B/C & 16A/B/C	82.0	-
Surveillance Materials		
Intermediate Shell 11-1	65.0	-
Surveillance Weld (Heat # 4P4784)	54.0	-

Notes:

(a) All values are based on Tables 1 and 2 of Regulatory Guide 1.99, Revision 2 (Position 1.1) using the Cu and Ni weight percent values given in Table 3-1 of this report. Dashes indicate when a category is not applicable to the material.

(b) Values are from Table 3-2 of this report.

4 PRESSURIZED THERMAL SHOCK

A limiting condition on RPV integrity known as Pressurized Thermal Shock (PTS) may occur during a severe system transient such as a loss-of-coolant accident (LOCA) or steam line break. Such transients may challenge the integrity of the RPV under the following conditions: severe overcooling of the inside surface of the vessel wall followed by high pressurization, significant degradation of vessel material toughness caused by radiation embrittlement, and the presence of a critical-size defect anywhere within the vessel wall.

In 1985, the USNRC issued a formal ruling on PTS (10 CFR 50.61 [Reference 6]) that established screening criteria on pressurized water reactor (PWR) vessel embrittlement, as measured by the maximum reference nil-ductility transition temperature in the limiting beltline component at the end of license, termed RT_{PTS}. RT_{PTS} screening values were set by the USNRC for beltline axial welds, forgings or plates, and for beltline circumferential weld seams for plant operation to the end of plant license. All domestic PWR vessels have been required to evaluate vessel embrittlement in accordance with the criteria through the end of license. The USNRC revised 10 CFR 50.61 in 1991 and 1995 to change the procedure for calculating radiation embrittlement. These revisions make the procedure for calculating the reference temperature for pressurized thermal shock (RT_{PTS}) values consistent with the methods given in Regulatory Guide 1.99, Revision 2 (Reference 8).

These accepted methods were used with the surface fluence values of Section 2 to calculate the following RT_{PTS} values for the VCSNS Unit 1 RPV materials. The end of SPEO RT_{PTS} calculations are presented in Table 4-1.

PTS Conclusion

All of the beltline and extended beltline materials in the VCSNS Unit 1 RV are below the RT_{PTS} screening criteria values of 270°F for base metal and/or longitudinal welds, and 300°F for circumferentially oriented welds through SPEO (72 EFPY). These RT_{PTS} values are based on the revised initial RT_{NDT} values in PWROG-21037-NP (Reference 7), which are developed using ASME Section III (Reference 11) and, if needed, NUREG-0800, BTP 5-3 (Reference 12) methodologies. Limiting fluence values corresponding to the lowest extent of the nozzle welds were used to calculate the RT_{PTS} values for both the nozzle welds and nozzle forgings.

The VCSNS Unit 1 limiting RT_{PTS} value for base metal and longitudinal welds at 72 EFPY is 152.5°F (see Table 4-1), which corresponds to VCSNS Unit 1 Intermediate Shell 11-1 based on Regulatory Guide 1.99, Position 1.1. Note, that there is surveillance data available for this material that indicated the ΔRT_{NDT} will be less than that predicted by RG 1.99, Position 1.1. However, because the surveillance data was determined to be non-conservative, it is not credited here. The VCSNS Unit 1 limiting RT_{PTS} value for circumferentially oriented welds at 72 EFPY is 42.5°F (see Table 4-1), which corresponds to the VCSNS Unit 1 Intermediate to Lower Shell Circumferential Weld Heat # 4P4784 based on Regulatory Guide 1.99, Position 2.1 with credible surveillance data. The credible surveillance data for Heat # 4P4784 supersedes the higher RT_{PTS} based on RG 1.99, Position 1.1. Note, both the Position 1.1 and 2.1 remain below 300°F.

Table 4-1 RT_{PTS} Calculations for VCSNS Unit 1 at 72 EFPY[(a)]

Material	R.G. 1.99, Rev. 2 Position	CF[(b)]	Surf. Fluence (x 10^{19} n/cm², E > 1.0MeV)[(c)]	Surf. FF[(d)]	$RT_{NDT(U)}$ (°F)	Predicted ΔRT_{NDT} (°F)[(e)]	σ_I (°F)	σ_Δ (°F)[(f)]	M (°F)	ART (°F)
Beltline Materials										
Intermediate Shell 11-1 (Heat # A9154-1)	1.1	65.0	9.04	1.501	21	97.5	0.0	17.0	34.0	*152.5*
Using non-credible surveillance data[(g)]	2.1	43.5	9.04	1.501	21	65.3	0.0	17.0	34.0	120.3
Intermediate Shell 11-2 (Heat # A9153-2)	1.1	58.0	9.04	1.501	-20	87.0	0.0	17.0	34.0	101.0
Lower Shell 10-1 (Heat # C9923-1)	1.1	51.0	9.06	1.501	5	76.6	0.0	17.0	34.0	115.6
Lower Shell 10-2 (Heat # C9923-2)	1.1	51.0	9.06	1.501	4	76.6	0.0	17.0	34.0	114.6
Intermediate Shell Long. Weld Seams BC & BD (Heat # 4P4784)	1.1	68.0	3.03	1.293	-49	87.9	0.0	28.0	56.0	94.9
Using credible surveillance data[(g)]	2.1	42.3	3.03	1.293	-49	54.7	0.0	14.0	28.0	33.7
Intermediate to Lower Shell Circ. Weld Seam AB (Heat # 4P4784)	1.1	68.0	9.04	1.501	-49	102.0	0.0	28.0	56.0	109.0
Using credible surveillance data[(g)]	2.1	42.3	9.04	1.501	-49	63.5	0.0	14.0	28.0	42.5
Lower Shell Long. Weld Seams BA & BB (Heat # 4P4784)	1.1	68.0	3.08	1.297	-49	88.2	0.0	28.0	56.0	95.2
Using credible surveillance data[(g)]	2.1	42.3	3.08	1.297	-49	54.9	0.0	14.0	28.0	33.9
Extended Beltline Materials										
Nozzle Shell 12-1 (Heat # C9955-2)	1.1	90.1	0.400	0.746	9	67.2	0.0	17.0	34.0	110.2
Nozzle Shell 12-2 (Heat # C0123-2)	1.1	82.6	0.400	0.746	15	61.6	0.0	17.0	34.0	110.6
Inlet Nozzle 436B-1 (Heat # Q2Q41W)	1.1	92.1	0.0310	0.224	-20	20.6	0.0	10.3	20.6	21.2
Inlet Nozzle 436B-2 (Heat # Q2Q39W)	1.1	93.0	0.0310	0.224	0	20.8	0.0	10.4	20.8	41.6
Inlet Nozzle 436B-3 (Heat # Q2Q39W)	1.1	93.0	0.0310	0.224	-20	20.8	0.0	10.4	20.8	21.6
Outlet Nozzle 437B-1 (Heat # Q2Q40)	1.1	93.0	0.0132	0.132	-10	12.3	0.0	6.1	12.3	14.5
Outlet Nozzle 437B-2 (Heat # Q2Q40W)	1.1	93.0	0.0132	0.132	-10	12.3	0.0	6.1	12.3	14.5
Outlet Nozzle 437B-3 (Heat # Q2Q44W)	1.1	92.6	0.0132	0.132	0	12.2	0.0	6.1	12.2	24.4
Nozzle to Intermediate Shell Circ. Weld Seam AC (Heat # 4P4784)	1.1	68.0	0.425	0.762	-49	51.8	0.0	25.9	51.8	54.7
Using credible surveillance data[(g)]	2.1	42.3	0.425	0.762	-49	32.2	0.0	14.0	28.0	11.2
Nozzle Shell Long. Weld Seams BE and BF	1.1	82.0	0.400	0.762	10	61.2	0.0	28.0	56.0	127.2

Table 4-1 RT_{PTS} Calculations for VCSNS Unit 1 at 72 EFPY[(a)]

Material	R.G. 1.99, Rev. 2 Position	CF[(b)]	Surf. Fluence (x 10^{19} n/cm^2, E > 1.0MeV)[(c)]	Surf. FF[(d)]	$RT_{NDT(U)}$ (°F)	Predicted ΔRT_{NDT} (°F)[(e)]	σ_I (°F)	σ_Δ (°F)[(f)]	M (°F)	ART (°F)
Inlet/Outlet Nozzle Forgings to Nozzle Shell Weld Seams 15A/B/C & 16A/B/C	1.1	82.0	0.0310	0.224	10	18.4	0.0	9.2	18.4	46.7

Notes:

(a) The 10 CFR 50.61 methodology was utilized in the calculation of the RT_{PTS} values.

(b) Chemistry factors are taken from Table 3-3.

(c) Fluence taken from Table 2-2 of this report.

(d) FF = fluence factor = $f^{(0.28 - 0.10*\log(f))}$.

(e) $RT_{NDT(U)}$ values taken from Table 3-1.

(f) Per 10 CFR 50.61, the base metal σ_Δ = 17°F when surveillance data are non-credible or not used to determine the CF, and the base metal σ_Δ = 8.5°F when credible surveillance data are used. Also, per 10 CFR 50.61, the weld metal σ_Δ = 28°F when surveillance data are non-credible or not used to determine the CF, and the weld metal σ_Δ = 14°F when credible surveillance data are used. However, σ_Δ need not exceed 0.5*ΔRT_{NDT} for either base metals or welds, with or without surveillance data.

(g) The credibility evaluation for the VCSNS Unit 1 surveillance data in Appendix A of this report determined that the VCSNS Unit 1 surveillance data for the Intermediate Shell 11-1 (Heat # A9154-1) is deemed non-credible and the Surveillance Weld (Heat # 4P4784) is deemed credible.

5 UPPER-SHELF ENERGY

The decrease in Charpy upper-shelf energy (USE) is associated with the determination of acceptable RPV toughness during the license renewal period when the vessel is exposed to additional irradiation.

The requirements on USE are included in 10 CFR 50, Appendix G (Reference 3). 10 CFR 50, Appendix G requires utilities to submit an analysis at least 3 years prior to the time that the USE of any RPV material is predicted to drop below 50 ft-lb, as measured by Charpy V-notch specimen testing.

There are two methods that can be used to predict the decrease in USE with irradiation, depending on the availability of credible surveillance capsule data as defined in Regulatory Guide 1.99, Revision 2 (Reference 8). For vessel beltline materials that are not in the surveillance program or have non-credible data, the Charpy USE (Position 1.2) is assumed to decrease as a function of fluence and copper content, as indicated in Regulatory Guide 1.99, Revision 2. When two or more credible surveillance sets become available from the reactor, they may be used to determine the Charpy USE of the surveillance material. The surveillance data are then used in conjunction with the Regulatory Guide to predict the change in USE (Position 2.2) of the RPV material due to irradiation. Per Regulatory Guide 1.99, Revision 2, when credible data exist, the Position 2.2 projected USE value should be used in preference to the Position 1.2 projected USE value. Note, if data from the surveillance materials is determined to be non-credible for determination of ΔRT_{NDT} by Credibility Criterion 3 of Regulatory Guide 1.99, Revision 2, then "they may be credible for determining decrease in upper-shelf energy if the upper shelf can be clearly determined, following the definition given in ASTM E 185-82."

The 72 EFPY Position 1.2 USE values of the vessel materials can be predicted using the corresponding 1/4T fluence projections, the copper content of the materials, and Figure 2 in Regulatory Guide 1.99, Revision 2 (see Figure 5-1).

The predicted Position 2.2 USE values are determined for the RV materials that are contained in the surveillance program by using the reduced plant surveillance data along with the corresponding 1/4T fluence projection. The surveillance data was plotted in Regulatory Guide 1.99, Revision 2, Figure 2 (see Figure 5-1) using the surveillance capsule fluence values documented in Table 2-1 of this report for VCSNS Unit 1. This data was fitted by drawing a line parallel to the existing lines as the upper bound of all the surveillance data. These reduced lines were used instead of the existing lines to determine the Position 2.2 end of SPEO USE values.

The projected USE values were calculated to determine if the VCSNS Unit 1 beltline and extended beltline materials remain above the 50 ft-lb criterion at 72 EFPY (end of SPEO). These calculations are summarized in Table 5-1.

USE Conclusion

As shown in Table 5-1, all VCSNS Unit 1 RV materials are projected to remain at or above the USE screening criterion value of 50 ft-lb at 72 EFPY. The limiting USE value at 72 EFPY is 63 ft-lb (see Table 5-1); this value corresponds to Intermediate Shell 11-1 using Position 2.2. The surveillance data for Intermediate Shell 11-1 is used despite it being determined to be non-credible, as the upper shelf can be clearly determined for the surveillance specimens (see WCAP-16298-NP). Note, both the Position 1.2 and 2.2 results for Intermediate Shell 11-1 remain above 50 ft-lb.

Table 5-1 Predicted USE Values at 72 EFPY for the VCSNS Unit 1 Beltline and Extended Beltline Materials

Material	Wt % Cu[a]	1/4T Fluence (x 10^{19} n/cm^2, E > 1.0 MeV)[b]	Unirradiated USE (ft-lb)[a]	Projected USE Decrease (%)	Projected USE (ft-lb)
Position 1.2[c]					
Intermediate Shell 11-1 (Heat # A9154-1)	0.10	5.68	76	29	54
Intermediate Shell 11-2 (Heat # A9153-2)	0.09	5.68	107	29	76
Lower Shell 10-1 (Heat # C9923-1)	0.08	5.69	106	29	75
Lower Shell 10-2 (Heat # C9923-2)	0.08	5.69	92	29	65
Intermediate Shell Long. Weld Seams BC & BD (Heat # 4P4784)	0.05	1.90	86	22	67
Intermediate to Lower Shell Circ. Weld Seam AB (Heat # 4P4784)	0.05	5.68	86	29	61
Lower Shell Long. Weld BA & BB (Heat # 4P4784)	0.05	1.93	86	23	66
Nozzle Shell 12-1 (Heat # C9955-2)	0.13	0.251	101	17	84
Nozzle Shell 12-2 (Heat # C0123-2)	0.12	0.251	91	16	76
Inlet Nozzle 436B-1 (Heat # Q2Q41W)	0.127	0.0310[d]	152	10	137
Inlet Nozzle 436B-2 (Heat # Q2Q39W)	0.127	0.0310[d]	115	10	104
Inlet Nozzle 436B-3 (Heat # Q2Q39W)	0.127	0.0310[d]	138	10	124
Outlet Nozzle 437B-1 (Heat # Q2Q40)	0.127	0.0132[d]	159	9	145
Outlet Nozzle 437B-2 (Heat # Q2Q40W)	0.127	0.0132[d]	165	9	150
Outlet Nozzle 437B-3 (Heat # Q2Q44W)	0.127	0.0132[d]	155	9	141
Nozzle to Intermediate Shell Circ. Weld Seam AC (Heat # 4P4784)	0.05	0.267	86	14	74
Nozzle Shell Long. Weld Seams BE & BF	0.06	0.251	80	15	68
Inlet/Outlet Nozzle Forgings to Nozzle Shell Weld Seams 15A/B/C & 16A/B/C	0.06	0.0310[d]	80	9	73
Position 2.2[c]					
Intermediate Shell 11-1 (Heat #A9154-1)	0.10	5.68	76	17	63
Intermediate Shell Long. Weld Seams BC & BD (Heat # 4P4784)	0.05	1.90	86	9	78
Intermediate to Lower Shell Circ. Weld Seam AB (Heat # 4P4784)	0.05	5.68	86	12	76
Lower Shell Long. Weld Seams BA & BB (Heat # 4P4784)	0.05	1.93	86	9	78
Nozzle to Intermediate Shell Circ. Weld Seam AC (Heat # 4P4784)	0.05	0.267	86	6	81

Notes contained on following page.

Notes:

(a) Copper weight percent values and unirradiated USE values were taken from Table 3-1 of this report. If the base metal or weld Cu weight percentages are below the minimum value presented in Figure 2 of Reg Guide 1.99 (0.1 for base metal and 0.05 for welds), then the Cu weight percentages were conservatively rounded up to the minimum value for projected USE decrease determination.

(b) Values taken from Table 6-2 of this report. Fluence values above 10^{17} n/cm^2 (E > 1.0 MeV) but below 2 x 10^{17} n/cm^2 (E > 1.0 MeV) were rounded to 2 x 10^{17} n/cm^2 (E > 1.0 MeV) when determining the % decrease because 2 x 10^{17} n/cm^2 is the lowest fluence displayed in Figure 2 of RG 1.99.

(c) Position 1.2 percentage USE decrease values were calculated by plotting the 1/4T fluence values on RG 1.99, Figure 2 and using the material-specific Cu wt. % values. The percent-loss lines were extended into the low fluence area of RG 1.99, Figure 2, i.e., below 10^{18} n/cm^2, in order to determine the USE % decrease, as needed. Position 2.2 percentage USE decrease values were determined by drawing an upper-bound line parallel to the existing RG 1.99, Figure 2 lines through the applicable surveillance data points. These results should be used in preference to the existing graph lines for determining the decrease in USE, because the surveillance data is credible.

(d) Values are the maximum fluence values instead of the 1/4T fluence values.

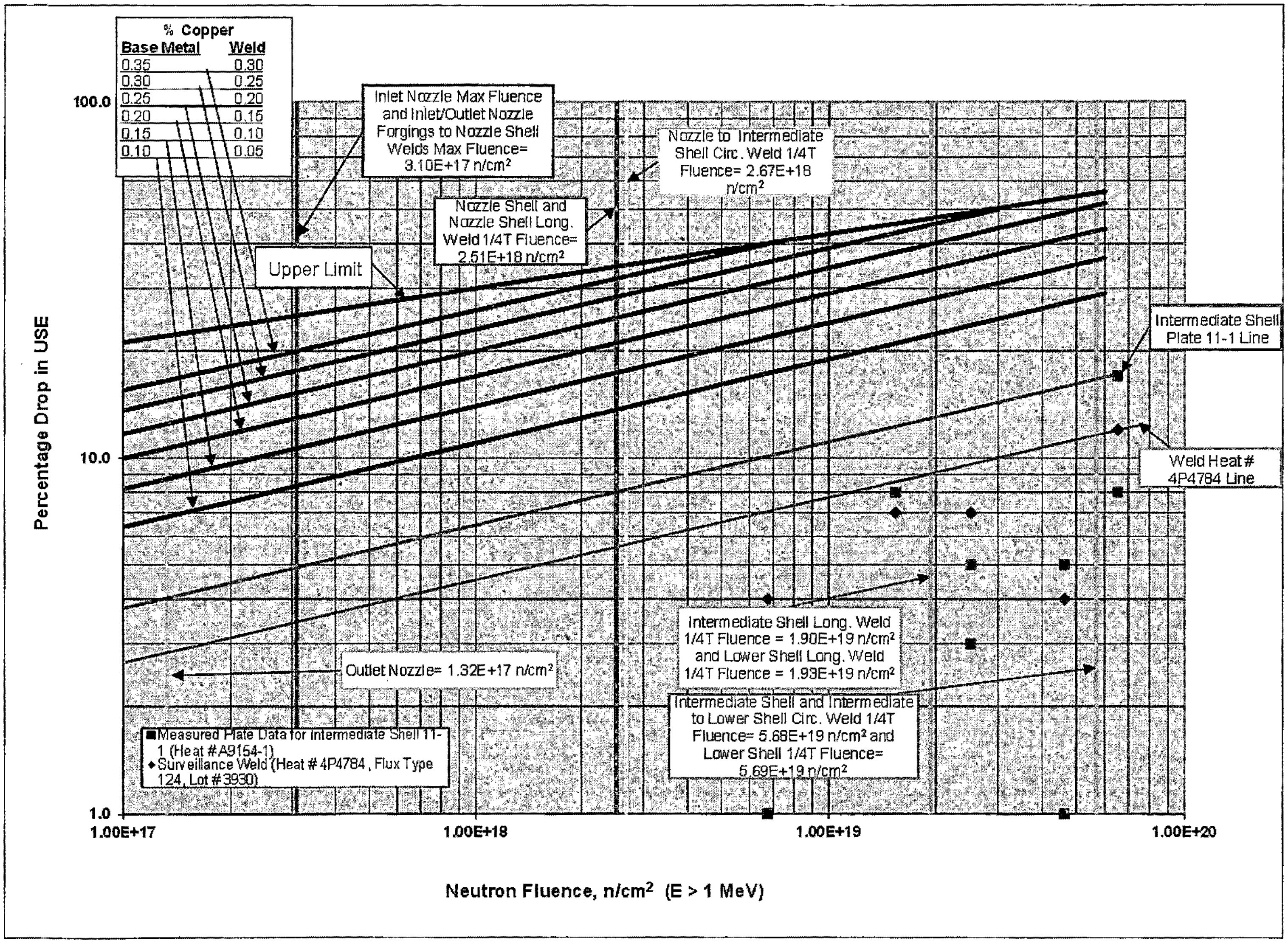

Figure 5-1 **Regulatory Guide 1.99, Revision 2, Position 1.2 & 2.2 Predicted Decrease in Upper-Shelf Energy as a Function of Copper and Fluence for VCSNS Unit 1 at the End of SPEO (72 EFPY)**

6 HEATUP AND COOLDOWN PRESSURE-TEMPERATURE LIMIT CURVES

Heatup and cooldown limit curves are calculated using the most limiting value of RT_{NDT} (reference nil-ductility transition temperature) corresponding to the limiting material in the beltline region of the RPV. The most limiting RT_{NDT} of the material in the core (beltline) region of the RPV is determined by using the unirradiated RPV material fracture toughness properties and estimating the irradiation-induced shift (ΔRT_{NDT}).

6.1 ADJUSTED REFERENCE TEMPERATURES CALCULATION

RT_{NDT} increases as the material is exposed to fast-neutron irradiation; therefore, to find the most limiting RT_{NDT} at any time period in the reactor's life, ΔRT_{NDT} due to the radiation exposure associated with that time period must be added to the original unirradiated RT_{NDT}. Using the adjusted reference temperature (ART) values, pressure-temperature (P-T) limit curves are determined in accordance with the requirements of 10 CFR Part 50, Appendix G (Reference 3), as augmented by Appendix G to Section XI of the ASME Boiler and Pressure Vessel Code (Reference 13).

The P-T limit curves for normal heatup and cooldown of the primary reactor coolant system for VCSNS Unit 1 were previously developed in WCAP-16035-NP (Reference 14). The existing P-T limit curves are based on the limiting beltline material ART values, which are influenced by both the fluence and the initial material properties of that material. Since the development of the curves, the fluence values and initial material properties used to calculate ART values have been updated and an applicability check of the current P-T limit curves is appropriate.

To confirm whether or not the current P-T limit curves will remain valid through the PEO and through the SPEO, updated ART values for the limiting materials were computed to account for updated 56 EFPY and 72 EFPY fluence values, updated Chemistry Factor values, and updated initial RT_{NDT} values. The Regulatory Guide 1.99, Revision 2 (Reference 8) methodology was used along with the surface fluence of Section 2 to calculate ART values, which are summarized in Table 6-3 through Table 6-8. Note, the inlet/outlet nozzle forgings and associated welds neglect attenuation through the material; thus, ART calculations are only needed at one location, i.e., the location of maximum fluence. Table 6-1 and Table 6-2 show the surface, 1/4T, and 3/4T fluence values for 56 EFPY and 72 EFPY, respectively.

ART projections contained herein are based on those projected fluence values with a 1.1 bias on the peripheral and re-entrant corner assembly relative powers.

Table 6-1 VCSNS Unit 1 Fluence and Fluence Factor Values for the Surface, 1/4T, and 3/4T Locations at 56 EFPY

Material Description	Surface Fluence[(a)] (x 10^{19} n/cm^2, E> 1.0 MeV)	Surface FF[(b)]	1/4T Fluence[(a)] (x 10^{19} n/cm^2, E > 1.0 MeV)	1/4T FF[(b)]	3/4T Fluence[(a)] (x 10^{19} n/cm^2, E> 1.0 MeV)	3/4T FF[(b)]
			Beltline			
Intermediate Shell 11-1 (Heat # A9154-1)	6.99	1.463	4.39	1.376	1.732	1.151
Intermediate Shell 11-2 (Heat # A9153-2)	6.99	1.463	4.39	1.376	1.732	1.151
Lower Shell 10-1 (Heat # C9923-1)	7.00	1.463	4.40	1.376	1.735	1.152
Lower Shell 10-2 (Heat # C9923-2)	7.00	1.463	4.40	1.376	1.735	1.152
Intermediate Shell Long. Weld (Heat # 4P4784)	2.35	1.231	1.48	1.108	0.582	0.849
Intermediate to Lower Shell Circ. Weld (Heat # 4P4784)	6.99	1.463	4.39	1.376	1.732	1.151
Lower Shell Long. Weld (Heat # 4P4784)	2.39	1.235	1.50	1.112	0.592	0.853
			Extended Beltline			
Nozzle Shell 12-1 (Heat # C9955-2)	0.309	0.678	0.194	0.562	0.0766	0.366
Nozzle Shell 12-2 (Heat # C0123-2)	0.309	0.678	0.194	0.562	0.0766	0.366
Inlet Nozzle 436B-1 (Heat # Q2Q41W)	0.0239	0.192	See Note (d)			
Inlet Nozzle 436B-2 (Heat # Q2Q39W)	0.0239	0.192	See Note (d)			
Inlet Nozzle 436B-3 (Heat # Q2Q39W)	0.0239	0.192	See Note (d)			
Outlet Nozzle 437B-1 (Heat # Q2Q40)	0.0102	0.111	See Note (d)			
Outlet Nozzle 437B-2 (Heat # Q2Q40W)	0.0102	0.111	See Note (d)			
Outlet Nozzle 437B-3 (Heat # Q2Q44W)	0.0102	0.111	See Note (d)			
Nozzle to Intermediate Shell Circ. Weld (Heat # 4P4784)	0.328	0.693	0.206	0.577	0.0813	0.377
Nozzle Shell Long. Welds[(c)]	0.309	0.678	0.194	0.562	0.0766	0.366
Inlet/Outlet Nozzle Forgings to Nozzle Shell Welds	0.0239	0.192	See note (d)			

Notes:

(a) The surface fluence values for the RV materials were determined by interpolation from data in Table 2-2. The 1/4T and 3/4T fluence values were calculated from the surface fluence, the RV beltline thickness (7.75 inches) and equation $f = f_{surf} * e^{-0.24(x)}$ from Regulatory Guide 1.99, Revision 2, where x = the depth into the vessel wall (inches).

(b) FF = fluence factor = $f^{(0.28 - 0.10*\log(f))}$.

(c) Exposure values for the nozzle shell longitudinal welds are bounded by the exposure values for the nozzle shell.

(d) Analysis of the nozzle forgings and associated welds are conservatively performed using the maximum fluence through the vessel wall.

Table 6-2 VCSNS Unit 1 Fluence and Fluence Factor Values for the Surface, 1/4T, and 3/4T Locations at 72 EFPY

Material Description	Surface Fluence(a) (x 10^{19} n/cm^2, E> 1.0 MeV)	Surface FF(b)	1/4T Fluence(a) (x 10^{19} n/cm^2, E > 1.0 MeV)	1/4T FF(b)	3/4T Fluence(a) (x 10^{19} n/cm^2, E> 1.0 MeV)	3/4T FF(b)
Beltline						
Intermediate Shell 11-1 (Heat # A9154-1)	9.04	1.501	5.68	1.427	2.240	1.218
Intermediate Shell 11-2 (Heat # A9153-2)	9.04	1.501	5.68	1.427	2.240	1.218
Lower Shell 10-1 (Heat # C9923-1)	9.06	1.501	5.69	1.427	2.245	1.219
Lower Shell 10-2 (Heat # C9923-2)	9.06	1.501	5.69	1.427	2.245	1.219
Intermediate Shell Long. Weld (Heat # 4P4784)	3.03	1.293	1.90	1.176	0.751	0.920
Intermediate to Lower Shell Circ. Weld (Heat # 4P4784)	9.04	1.501	5.68	1.427	2.240	1.218
Lower Shell Long. Weld (Heat # 4P4784)	3.08	1.297	1.93	1.180	0.763	0.924
Extended Beltline						
Nozzle Shell 12-1 (Heat # C9955-2)	0.400	0.746	0.251	0.625	0.0991	0.415
Nozzle Shell 12-2 (Heat # C0123-2)	0.400	0.746	0.251	0.625	0.0991	0.415
Inlet Nozzle 436B-1 (Heat # Q2Q41W)	0.0310	0.224	See Note (d)			
Inlet Nozzle 436B-2 (Heat # Q2Q39W)	0.0310	0.224	See Note (d)			
Inlet Nozzle 436B-3 (Heat # Q2Q39W)	0.0310	0.224	See Note (d)			
Outlet Nozzle 437B-1 (Heat # Q2Q40)	0.0132	0.132	See Note (d)			
Outlet Nozzle 437B-2 (Heat # Q2Q40W)	0.0132	0.132	See Note (d)			
Outlet Nozzle 437B-3 (Heat # Q2Q44W)	0.0132	0.132	See Note (d)			
Nozzle to Intermediate Shell Circ. Weld (Heat # 4P4784)	0.425	0.762	0.267	0.640	0.105	0.427
Nozzle Shell Long. Welds(c)	0.400	0.746	0.251	0.625	0.099	0.415
Inlet/Outlet Nozzle Forgings to Nozzle Shell Welds	0.0310	0.224	See note (d)			

Notes:

(a) The surface fluence values for the RV materials were determined from Table 2-2. The 1/4T and 3/4T fluence values were calculated from the surface fluence, the RV beltline thickness (7.75 inches) and equation $f = f_{surf} * e^{-0.24\,(x)}$ from Regulatory Guide 1.99, Revision 2, where x = the depth into the vessel wall (inches).

(b) FF = fluence factor = $f^{(0.28 - 0.10*\log(f))}$.

(c) Exposure values for the nozzle shell longitudinal welds are bounded by the exposure values for the nozzle shell.

(d) Analysis of the nozzle forgings and associated welds are conservatively performed using the maximum fluence through the vessel wall.

Table 6-3 Calculation of the VCSNS Unit 1 ART Values at the 1/4T Location for the Reactor Vessel Beltline and Extended Beltline Materials at the End of PEO (56 EFPY)[(a)]

Material	R.G. 1.99, Rev. 2 Position	CF[(b)]	1/4T Fluence (x 10^{19} n/cm^2, E > 1.0 MeV)[(c)]	1/4T FF[(d)]	$RT_{NDT(U)}$ (°F)[(e)]	Predicted ΔRT_{NDT} (°F)	σ_I (°F)	σ_Δ (°F)[(f)]	M (°F)	ART (°F)
Beltline Materials										
Intermediate Shell 11-1 (Heat # A9154-1)	1.1	65.0	4.39	1.376	21	89.4	0.0	17.0	34.0	144.4
Using non-credible surveillance data[(g)]	2.1	43.5	4.39	1.376	21	59.9	0.0	17.0	34.0	114.9
Intermediate Shell 11-2 (Heat # A9153-2)	1.1	58.0	4.39	1.376	-20	79.8	0.0	17.0	34.0	93.8
Lower Shell 10-1 (Heat # C9923-1)	1.1	51.0	4.40	1.376	5	70.2	0.0	17.0	34.0	109.2
Lower Shell 10-2 (Heat # C9923-2)	1.1	51.0	4.40	1.376	4	70.2	0.0	17.0	34.0	108.2
Intermediate Shell Long. Weld Seams BC & BD (Heat # 4P4784)	1.1	68.0	1.48	1.108	-49	75.3	0.0	28.0	56.0	82.3
Using credible surveillance data[(g)]	2.1	42.3	1.48	1.108	-49	46.9	0.0	14.0	28.0	25.9
Intermediate to Lower Shell Circ. Weld Seam AB (Heat # 4P4784)	1.1	68.0	4.39	1.376	-49	93.6	0.0	28.0	56.0	100.6
Using credible surveillance data[(g)]	2.1	42.3	4.39	1.376	-49	58.2	0.0	14.0	28.0	37.2
Lower Shell Long. Weld Seams BA & BB (Heat # 4P4784)	1.1	68.0	1.50	1.112	-49	75.6	0.0	28.0	56.0	82.6
Using credible surveillance data[(g)]	2.1	42.3	1.50	1.112	-49	47.1	0.0	14.0	28.0	26.1
Extended Beltline Materials										
Nozzle Shell 12-1 (Heat # C9955-2)	1.1	90.1	0.194	0.563	9	50.7	0.0	17.0	34.0	93.7
Nozzle Shell 12-2 (Heat # C0123-2)	1.1	82.6	0.194	0.563	15	46.5	0.0	17.0	34.0	95.5
Nozzle to Intermediate Shell Circ. Weld Seam AC (Heat # 4P4784)	1.1	68.0	0.206	0.577	-49	39.2	0.0	19.6	39.2	29.4
Using credible surveillance data[(g)]	2.1	42.3	0.206	0.577	-49	24.4	0.0	12.2	24.4	-0.2
Nozzle Shell Long. Weld Seams BE and BF	1.1	82.0	0.194	0.563	10	46.1	0.0	23.1	46.1	102.3

Notes contained on following page.

Notes:

(a) The Regulatory Guide 1.99, Revision 2 methodology was utilized in the calculation of the ART values.

(b) Chemistry factors are taken from Table 3-3.

(c) Fluence taken from Table 6-1 of this report.

(d) FF = fluence factor = $f^{(0.28 - 0.10*\log(f))}$.

(e) $RT_{NDT(U)}$ (Unirradiated RT_{NDT}) values taken from Table 3-1.

(f) Per the guidance of Regulatory Guide 1.99, Revision 2, the base metal σ_Δ = 17°F for Position 1.1 and Position 2.1 with non-credible surveillance data, and the base metal σ_Δ = 8.5°F for Position 2.1 with credible surveillance data. Also, per Regulatory Guide 1.99, Revision 2, the weld metal σ_Δ = 28°F for Position 1.1 and Position 2.1 with non-credible surveillance data, and the weld metal σ_Δ = 14°F for Position 2.1 with credible surveillance data. However, σ_Δ need not exceed $0.5*\Delta RT_{NDT}$ for either base metals or welds, with or without surveillance data.

(g) The credibility evaluation for the VCSNS Unit 1 surveillance data in Appendix A determined that the VCSNS Unit 1 surveillance data for the Intermediate Shell 11-1 (Heat # A9154-1) is deemed non-credible and the Surveillance Weld (Heat # 4P4784) is deemed credible.

Table 6-4 Calculation of the VCSNS Unit 1 ART Values at the 3/4T Location for the Reactor Vessel Beltline and Extended Beltline Materials at the End of PEO (56 EFPY)[(a)]

Material	R.G. 1.99, Rev. 2 Position	CF[(b)]	3/4T Fluence (x 10^{19} n/cm², E > 1.0 MeV)[(c)]	3/4T FF[(d)]	$RT_{NDT(U)}$ (°F)[(e)]	Predicted ΔRT_{NDT} (°F)	σ_I (°F)	σ_Δ (°F)[(f)]	M (°F)	ART (°F)
Beltline Materials										
Intermediate Shell 11-1 (Heat # A9154-1)	1.1	65.0	1.73	1.151	21	74.8	0.0	17.0	34.0	129.8
Using non-credible surveillance data[(g)]	2.1	43.5	1.73	1.151	21	50.1	0.0	17.0	34.0	105.1
Intermediate Shell 11-2 (Heat # A9153-2)	1.1	58.0	1.73	1.151	-20	66.8	0.0	17.0	34.0	80.8
Lower Shell 10-1 (Heat # C9923-1)	1.1	51.0	1.73	1.152	5	58.7	0.0	17.0	34.0	97.7
Lower Shell 10-2 (Heat # C9923-2)	1.1	51.0	1.73	1.152	4	58.7	0.0	17.0	34.0	96.7
Intermediate Shell Long. Weld Seams BC & BD (Heat # 4P4784)	1.1	68.0	0.582	0.849	-49	57.7	0.0	28.0	56.0	64.7
Using credible surveillance data[(g)]	2.1	42.3	0.582	0.849	-49	35.9	0.0	14.0	28.0	14.9
Intermediate to Lower Shell Circ. Weld Seam AB (Heat # 4P4784)	1.1	68.0	1.73	1.151	-49	78.3	0.0	28.0	56.0	85.3
Using credible surveillance data[(g)]	2.1	42.3	1.73	1.151	-49	48.7	0.0	14.0	28.0	27.7
Lower Shell Long. Weld Seams BA & BB (Heat # 4P4784)	1.1	68.0	0.592	0.853	-49	58.0	0.0	28.0	56.0	65.0
Using credible surveillance data[(g)]	2.1	42.3	0.592	0.853	-49	36.1	0.0	14.0	28.0	15.1
Extended Beltline Materials										
Nozzle Shell 12-1 (Heat # C9955-2)	1.1	90.1	0.0767	0.366	9	33.0	0.0	16.5	33.0	74.9
Nozzle Shell 12-2 (Heat # C0123-2)	1.1	82.6	0.0767	0.366	15	30.2	0.0	15.1	30.2	75.4
Nozzle to Intermediate Shell Circ. Weld Seam AC (Heat # 4P4784)	1.1	68.0	0.0814	0.377	-49	25.6	0.0	12.8	25.6	2.3
Using credible surveillance data[(g)]	2.1	42.3	0.0814	0.377	-49	15.9	0.0	8.0	15.9	-17.1
Nozzle Shell Long. Weld Seams BE and BF	1.1	82.0	0.0767	0.366	10	30.0	0.0	15.0	30.0	70.0

Notes contained on following page.

Notes:

(a) The Regulatory Guide 1.99, Revision 2 methodology was utilized in the calculation of the ART values.

(b) Chemistry factors are taken from Table 3-3.

(c) Fluence taken from Table 6-1 of this report.

(d) FF = fluence factor = $f^{(0.28 - 0.10*\log(f))}$.

(e) $RT_{NDT(U)}$ (Unirradiated RT_{NDT}) values taken from Table 3-1.

(f) Per the guidance of Regulatory Guide 1.99, Revision 2, the base metal σ_Δ = 17°F for Position 1.1 and Position 2.1 with non-credible surveillance data, and the base metal σ_Δ = 8.5°F for Position 2.1 with credible surveillance data. Also, per Regulatory Guide 1.99, Revision 2, the weld metal σ_Δ = 28°F for Position 1.1 and Position 2.1 with non-credible surveillance data, and the weld metal σ_Δ = 14°F for Position 2.1 with credible surveillance data. However, σ_Δ need not exceed $0.5*\Delta RT_{NDT}$ for either base metals or welds, with or without surveillance data.

(g) The credibility evaluation for the VCSNS Unit 1 surveillance data in Appendix A determined that the VCSNS Unit 1 surveillance data for the Intermediate Shell 11-1 (Heat # A9154-1) is deemed non-credible and the Surveillance Weld (Heat # 4P4784) is deemed credible.

Table 6-5 Calculation of the VCSNS Unit 1 ART Values for the Reactor Vessel Extended Beltline Nozzle Materials at the End of PEO (56 EFPY)[a]

Material	R.G. 1.99, Rev. 2 Position	CF[b]	Maximum Fluence (x 10^{19} n/cm^2, E > 1.0 MeV)[c]	Max FF[d]	$RT_{NDT(U)}$ (°F)[e]	Predicted ΔRT_{NDT} (°F)	σ_I (°F)	σ_Δ (°F)[f]	M (°F)	ART (°F)
Inlet Nozzle 436B-1 (Heat # Q2Q41W)	1.1	92.1	0.0239	0.192	-20	17.7	0.0	8.8	17.7	15.4
Inlet Nozzle 436B-2 (Heat # Q2Q39W)	1.1	93.0	0.0239	0.192	0	17.8	0.0	8.9	17.8	35.7
Inlet Nozzle 436B-3 (Heat # Q2Q39W)	1.1	93.0	0.0239	0.192	-20	17.8	0.0	8.9	17.8	15.7
Outlet Nozzle 437B-1 (Heat # Q2Q40)	1.1	93.0	0.0102	0.111	-10	10.3	0.0	5.2	10.3	10.6
Outlet Nozzle 437B-2 (Heat # Q2Q40W)	1.1	93.0	0.0102	0.111	-10	10.3	0.0	5.2	10.3	10.6
Outlet Nozzle 437B-3 (Heat # Q2Q44W)	1.1	92.6	0.0102	0.111	0	10.3	0.0	5.1	10.3	20.5
Inlet/Outlet Nozzle Forgings to Nozzle Shell Weld Seams 15A/B/C & 16A/B/C	1.1	82.0	0.0239	0.192	10	15.7	0.0	7.9	15.7	41.5

Notes:

(a) The Regulatory Guide 1.99, Revision 2 methodology was utilized in the calculation of the ART values.

(b) Chemistry factors are taken from Table 3-3.

(c) Fluence taken from Table 6-1 of this report.

(d) FF = fluence factor = $f^{(0.28 - 0.10*\log(f))}$.

(e) $RT_{NDT(U)}$ (Unirradiated RT_{NDT}) values taken from Table 3-1.

(f) Per the guidance of Regulatory Guide 1.99, Revision 2, the base metal σ_Δ = 17°F for Position 1.1 and Position 2.1 with non-credible surveillance data, and the base metal σ_Δ = 8.5°F for Position 2.1 with credible surveillance data. Also, per Regulatory Guide 1.99, Revision 2, the weld metal σ_Δ = 28°F for Position 1.1 and Position 2.1 with non-credible surveillance data, and the weld metal σ_Δ = 14°F for Position 2.1 with credible surveillance data. However, σ_Δ need not exceed 0.5*ΔRT_{NDT} for either base metals or welds, with or without surveillance data.

Table 6-6 Calculation of the VCSNS Unit 1 ART Values at the 1/4T Location for the Reactor Vessel Beltline and Extended Beltline Materials at the End of SPEO (72 EFPY)[(a)]

Material	R.G. 1.99, Rev. 2 Position	CF[(b)]	1/4T Fluence (x 10^{19} n/cm^2, E > 1.0 MeV)[(c)]	1/4T FF[(d)]	$RT_{NDT(U)}$ (°F)[(e)]	Predicted ΔRT_{NDT} (°F)	σ_I (°F)	σ_Δ (°F)[(f)]	M (°F)	ART (°F)
Beltline Materials										
Intermediate Shell 11-1 (Heat # A9154-1)	1.1	65.0	5.68	1.427	21	92.7	0.0	17.0	34.0	147.7
Using non-credible surveillance data[(g)]	2.1	43.5	5.68	1.427	21	62.1	0.0	17.0	34.0	117.1
Intermediate Shell 11-2 (Heat # A9153-2)	1.1	58.0	5.68	1.427	-20	82.7	0.0	17.0	34.0	96.7
Lower Shell 10-1 (Heat # C9923-1)	1.1	51.0	5.69	1.427	5	72.8	0.0	17.0	34.0	111.8
Lower Shell 10-2 (Heat # C9923-2)	1.1	51.0	5.69	1.427	4	72.8	0.0	17.0	34.0	110.8
Intermediate Shell Long. Weld Seams BC & BD (Heat # 4P4784)	1.1	68.0	1.90	1.176	-49	80.0	0.0	28.0	56.0	87.0
Using credible surveillance data[(g)]	2.1	42.3	1.90	1.176	-49	49.7	0.0	14.0	28.0	28.7
Intermediate to Lower Shell Circ. Weld Seam AB (Heat # 4P4784)	1.1	68.0	5.68	1.427	-49	97.0	0.0	28.0	56.0	104.0
Using credible surveillance data[(g)]	2.1	42.3	5.68	1.427	-49	60.3	0.0	14.0	28.0	39.3
Lower Shell Long. Weld Seams BA & BB (Heat # 4P4784)	1.1	68.0	1.93	1.180	-49	80.3	0.0	28.0	56.0	87.3
Using credible surveillance data[(g)]	2.1	42.3	1.93	1.180	-49	49.9	0.0	14.0	28.0	28.9
Extended Beltline Materials										
Nozzle Shell 12-1 (Heat # C9955-2)	1.1	90.1	0.251	0.625	9	56.3	0.0	17.0	34.0	99.3
Nozzle Shell 12-2 (Heat # C0123-2)	1.1	82.6	0.251	0.625	15	51.6	0.0	17.0	34.0	100.6
Nozzle to Intermediate Shell Circ. Weld Seam AC (Heat # 4P4784)	1.1	68.0	0.267	0.640	-49	43.6	0.0	21.8	43.6	38.1
Using credible surveillance data[(g)]	2.1	42.3	0.267	0.640	-49	27.1	0.0	13.5	27.1	5.2
Nozzle Shell Long. Weld Seams BE and BF	1.1	82.0	0.251	0.625	10	51.3	0.0	25.6	51.3	112.5

Notes contained on following page.

Notes:

(a) The Regulatory Guide 1.99, Revision 2 methodology was utilized in the calculation of the ART values.

(b) Chemistry factors are taken from Table 3-3.

(c) Fluence and Fluence Factors taken from Table 6-2 of this report.

(d) FF = fluence factor = $f^{(0.28 - 0.10*\log(f))}$.

(e) $RT_{NDT(U)}$ (Unirradiated RT_{NDT}) values taken from Table 3-1.

(f) Per the guidance of Regulatory Guide 1.99, Revision 2, the base metal σ_Δ = 17°F for Position 1.1 and Position 2.1 with non-credible surveillance data, and the base metal σ_Δ = 8.5°F for Position 2.1 with credible surveillance data. Also, per Regulatory Guide 1.99, Revision 2, the weld metal σ_Δ = 28°F for Position 1.1 and Position 2.1 with non-credible surveillance data, and the weld metal σ_Δ = 14°F for Position 2.1 with credible surveillance data. However, σ_Δ need not exceed $0.5*\Delta RT_{NDT}$ for either base metals or welds, with or without surveillance data.

(g) The credibility evaluation for the VCSNS Unit 1 surveillance data in Appendix A determined that the VCSNS Unit 1 surveillance data for the Intermediate Shell 11-1 (Heat # A9154-1) is deemed non-credible and the Surveillance Weld (Heat # 4P4784) is deemed credible.

Table 6-7 Calculation of the VCSNS Unit 1 ART Values at the 3/4T Location for the Reactor Vessel Beltline and Extended Beltline Materials at the End of SPEO (72 EFPY)[(a)]

Material	R.G. 1.99, Rev. 2 Position	CF[(b)]	3/4T Fluence (x 10^{19} n/cm^2, E > 1.0 MeV)[(c)]	3/4T FF[(d)]	$RT_{NDT(U)}$ (°F)[(e)]	Predicted ΔRT_{NDT} (°F)	σ_I (°F)	σ_Δ (°F)[(f)]	M (°F)	ART (°F)
Beltline Materials										
Intermediate Shell 11-1 (Heat # A9154-1)	1.1	65.0	2.24	1.218	21	79.2	0.0	17.0	34.0	134.2
Using non-credible surveillance data[(g)]	2.1	43.5	2.24	1.218	21	53.0	0.0	17.0	34.0	108.0
Intermediate Shell 11-2 (Heat # A9153-2)	1.1	58.0	2.24	1.218	-20	70.7	0.0	17.0	34.0	84.7
Lower Shell 10-1 (Heat # C9923-1)	1.1	51.0	2.25	1.219	5	62.2	0.0	17.0	34.0	101.2
Lower Shell 10-2 (Heat # C9923-2)	1.1	51.0	2.25	1.219	4	62.2	0.0	17.0	34.0	100.2
Intermediate Shell Long. Weld Seams BC & BD (Heat # 4P4784)	1.1	68.0	0.751	0.920	-49	62.5	0.0	28.0	56.0	69.5
Using credible surveillance data[(g)]	2.1	42.3	0.751	0.920	-49	38.9	0.0	14.0	28.0	17.9
Intermediate to Lower Shell Circ. Weld Seam AB (Heat # 4P4784)	1.1	68.0	2.24	1.218	-49	82.9	0.0	28.0	56.0	89.9
Using credible surveillance data[(g)]	2.1	42.3	2.24	1.218	-49	51.5	0.0	14.0	28.0	30.5
Lower Shell Long. Weld Seams BA & BB (Heat # 4P4784)	1.1	68.0	0.763	0.924	-49	62.8	0.0	28.0	56.0	69.8
Using credible surveillance data[(g)]	2.1	42.3	0.763	0.924	-49	39.1	0.0	14.0	28.0	18.1
Extended Beltline Materials										
Nozzle Shell 12-1 (Heat # C9955-2)	1.1	90.1	0.0991	0.415	9	37.4	0.0	17.0	34.0	80.4
Nozzle Shell 12-2 (Heat # C0123-2)	1.1	82.6	0.0991	0.415	15	34.3	0.0	17.0	34.0	83.3
Nozzle to Intermediate Shell Circ. Weld Seam AC (Heat # 4P4784)	1.1	68.0	0.105	0.427	-49	29.1	0.0	14.5	29.1	9.1
Using credible surveillance data[(g)]	2.1	42.3	0.105	0.427	-49	18.1	0.0	9.0	18.1	-12.8
Nozzle Shell Long. Weld Seams BE and BF	1.1	82.0	0.0991	0.415	10	34.0	0.0	17.0	34.0	78.1

Notes contained on following page.

Notes:

(a) The Regulatory Guide 1.99, Revision 2 methodology was utilized in the calculation of the ART values.

(b) Chemistry factors are taken from Table 3-3.

(c) Fluence taken from Table 6-2 of this report.

(d) FF = fluence factor = $f^{(0.28 - 0.10*\log(f))}$.

(e) $RT_{NDT(U)}$ (Unirradiated RT_{NDT}) values taken from Table 3-1.

(f) Per the guidance of Regulatory Guide 1.99, Revision 2, the base metal σ_Δ = 17°F for Position 1.1 and Position 2.1 with non-credible surveillance data, and the base metal σ_Δ = 8.5°F for Position 2.1 with credible surveillance data. Also, per Regulatory Guide 1.99, Revision 2, the weld metal σ_Δ = 28°F for Position 1.1 and Position 2.1 with non-credible surveillance data, and the weld metal σ_Δ = 14°F for Position 2.1 with credible surveillance data. However, σ_Δ need not exceed $0.5*\Delta RT_{NDT}$ for either base metals or welds, with or without surveillance data.

(g) The credibility evaluation for the VCSNS Unit 1 surveillance data in Appendix A determined that the VCSNS Unit 1 surveillance data for the Intermediate Shell 11-1 (Heat # A9154-1) is deemed non-credible and the Surveillance Weld (Heat # 4P4784) is deemed credible.

Table 6-8 Calculation of the VCSNS Unit 1 ART Values for the Reactor Vessel Extended Beltline Nozzle Materials at the End of SPEO (72 EFPY)[(a)]

Material	R.G. 1.99, Rev. 2 Position	CF[(b)]	Maximum Fluence (x 10^{19} n/cm^2, E > 1.0 MeV)[(c)]	Max FF[(d)]	$RT_{NDT(U)}$ (°F)[(e)]	Predicted ΔRT_{NDT} (°F)	σ_I (°F)	σ_Δ (°F)[(f)]	M (°F)	ART (°F)
Inlet Nozzle 436B-1 (Heat # Q2Q41W)	1.1	92.1	0.0310	0.224	-20	20.6	0.0	10.3	20.6	21.2
Inlet Nozzle 436B-2 (Heat # Q2Q39W)	1.1	93.0	0.0310	0.224	0	20.8	0.0	10.4	20.8	41.6
Inlet Nozzle 436B-3 (Heat # Q2Q39W)	1.1	93.0	0.0310	0.224	-20	20.8	0.0	10.4	20.8	21.6
Outlet Nozzle 437B-1 (Heat # Q2Q40)	1.1	93.0	0.0132	0.132	-10	12.3	0.0	6.1	12.3	14.5
Outlet Nozzle 437B-2 (Heat # Q2Q40W)	1.1	93.0	0.0132	0.132	-10	12.3	0.0	6.1	12.3	14.5
Outlet Nozzle 437B-3 (Heat # Q2Q44W)	1.1	92.6	0.0132	0.132	0	12.2	0.0	6.1	12.2	24.4
Inlet/Outlet Nozzle Forgings to Nozzle Shell Weld Seams 15A/B/C & 16A/B/C	1.1	82.0	0.0310	0.224	10	18.4	0.0	9.2	18.4	46.7

Notes:

(a) The Regulatory Guide 1.99, Revision 2 methodology was utilized in the calculation of the ART values.

(b) Chemistry factors are taken from Table 3-3.

(c) Fluence taken from Table 6-2 of this report.

(d) FF = fluence factor = $f^{(0.28 - 0.10*\log(f))}$.

(e) $RT_{NDT(U)}$ (Unirradiated RT_{NDT}) values taken from Table 3-1.

(f) Per the guidance of Regulatory Guide 1.99, Revision 2, the base metal σ_Δ = 17°F for Position 1.1 and Position 2.1 with non-credible surveillance data, and the base metal σ_Δ = 8.5°F for Position 2.1 with credible surveillance data. Also, per Regulatory Guide 1.99, Revision 2, the weld metal σ_Δ = 28°F for Position 1.1 and Position 2.1 with non-credible surveillance data, and the weld metal σ_Δ = 14°F for Position 2.1 with credible surveillance data. However, σ_Δ need not exceed 0.5*ΔRT_{NDT} for either base metals or welds, with or without surveillance data.

6.2 P-T LIMIT CURVES APPLICABILITY

This section determines the applicability term of the end of PEO P-T limit curves by comparing the ART values contained in the analysis of record (AOR) with the ART values calculated using the updated fluence projections and materials information contained herein. If the ART values used in the previous analysis are *higher* or *equal* to the ART values calculated using the updated fluence and material properties, then the applicability term of the current curves will remain unchanged. If the ART values used in the previous analysis are *lower* than the ART values calculated using the updated fluence and material properties, then the applicability term of the current curves may need to be shortened. This new period of applicability can be calculated based on a comparison of the ART values and linear interpolation using the fluence projections. P-T limit curves for the end of SPEO (72 EFPY) do not need to be submitted as part of the VCSNS Unit 1 Subsequent License Renewal Application since P-T limit curves are available as a part of the current license. However, new P-T limit curve development or an extension of the applicability of the current curves must be completed prior to the expiration of the current curves as specified in the VCSNS Unit 1 licensing basis.

Table 6-3 through Table 6-8 calculates the beltline and extended beltline ART values for VCSNS Unit 1 at the end of PEO (56 EFPY) and the end of SPEO (72 EFPY). The limiting end of SPEO ART values correspond to the Intermediate Shell 11-1.

Table 6-9 compares the TLAA limiting ART values at the end of PEO and the end of SPEO to the limiting ART values used in development of the existing PEO P-T limit curves implemented in the Technical Specifications which are based on WCAP-16305-NP (Reference 14).

Table 6-9 Summary of the Limiting ART Values

Vessel Wall Location	Limiting ART[b] (°F)		
	P-T Limit Curves AOR[a]	56 EFPY	72 EFPY
1/4T	153	144.4	147.7
3/4T	138	129.8	134.2

Notes:

(a) Information taken from WCAP-16305-NP.

(b) The limiting material (Intermediate Shell 11-1) corresponds to an axial flaw which is more limiting than a circumferential flaw.

Table 6-9 shows that the end of SPEO ART values at the 1/4T and 3/4T locations remain bounded by the ART values used in the current P-T limit curves. Thus, the P-T limit curves implemented in the VCSNS Unit 1 Technical Specifications will remain valid through the end of SPEO (72 EFPY) for the cylindrical shell materials. The extension of the P-T limit curve is allowed because the redefinition of the Intermediate Shell 11-1 (Heat # A9154-1) initial RT_{NDT} in PWROG-21037-NP resulted in a 9°F reduction.

Note that the terms of applicability for the P-T limits also implicitly confirm the bolt up temperature and flange temperature limits. The bolt up temperature and flange-notch temperature limit are not affected by embrittlement; thus, they are unaffected by license renewal and may remain the same. Since development of the P-T limit curves in WCAP-16305-NP, the closure head has been replaced. However, the RT_{NDT} of the replacement closure head is lower than the original head flange; thus, the P-T limit curves are not negatively affected.

Inlet and Outlet Nozzles P-T Limit Curves

NRC Regulatory Issue Summary (RIS) 2014-11 (Reference 22) requires that the P-T limit curves account for the higher stresses in the nozzle corner region due to the potential for more restrictive P-T limits, even if the RT_{NDT} for these components are not as high as those of the reactor vessel beltline shell materials that have simpler geometries. As shown in Table 2-2, the 80-year fluence at the inlet/outlet nozzle postulated 1/4T flaw is below the fluence threshold of RIS 2014-11, 1 x 10^{17} n/cm^2 (E > 1.0 MeV). In addition, PWROG-15109-NP-A (Reference 9) addresses the concern of higher stresses in nozzle corner regions generically for the U.S. PWR operating fleet. Therefore, the inlet/outlet nozzles are confirmed not to be limiting.

Conclusion

Based on the end of SPEO ART values calculated herein and the P-T limits analysis completed in WCAP-16305-NP (Reference 14), the P-T limit curves currently in the Technical Specifications will remain valid through 72 EFPY.

7 SURVEILLANCE CAPSULE WITHDRAWAL SCHEDULES

This section provides recommended capsule withdrawal schedules for VCSNS Unit 1 as well as technical justifications and demonstration of the schedule's compliance with ASTM E185-82 (Reference 15), as prescribed by 10 CFR 50, Appendix H (Reference 16) and consistent with the guidance of NUREG-1801, Revision 2 (GALL [Reference 17]) and NUREG-2191 (GALL-SLR [Reference 18]).

10 CFR 50, Appendix H states:

> *The design of the surveillance program and the withdrawal schedule must meet the requirements of the edition of ASTM E 185 that is current on the issue date of the ASME Code to which the reactor vessel was purchased; for reactor vessels purchased after 1982, the design of the surveillance program and the withdrawal schedule must meet the requirements of ASTM E 185–82. For reactor vessels purchased in or before 1982, later editions of ASTM E 185 may be used, but including only those editions through 1982. For each capsule withdrawal, the test procedures and reporting requirements must meet the requirements of ASTM E 185 to the extent practicable for the configuration of the specimens in the capsule.*

The VCSNS Unit 1 RV was designed and constructed to ASME Section III, 1971 Edition. Thus, per 10 CFR 50, Appendix H, the VCSNS Unit 1 surveillance program withdrawal schedule may meet the requirements of any version of the ASTM E185 standard from the 1970 version (which was current on the issue date of the ASME Code to which the RV was purchased) through the 1982 version. Per WCAP-9234 (Reference 19), the VCSNS Unit 1 surveillance capsule program was designed to the ASTM E185-73 (Reference 20) standard, which was the version active at that time. Therefore, the requirements of 10 CFR 50, Appendix H were met at the time of the design of the RV surveillance program. Figure 7-1 shows the initial installation of the capsules in the VCSNS Unit 1 RV.

Since that time VCSNS Unit 1 has implemented the capsule withdrawal schedules in Final Safety Analysis Report (FSAR) (Reference 21) to support license renewal (LR) to 60 years and to meet the requirements of ASTM E185-82 (Reference 15). Recommended changes to this surveillance capsule withdrawal schedule to support SPEO is provided in Table 7-1. This schedule meets the recommendations of ASTM E185-82 as required by 10 CFR 50, Appendix H and satisfies the guidance contained in the GALL and GALL-SLR. Specific details are described herein.

The first step in determining the surveillance capsule withdrawal schedule compliance is to determine the minimum number of capsules to withdraw and/or test. ASTM E185-82 bases the number of capsules on the maximum ΔRT_{NDT} projected at the vessel surface for all RV materials. Per Table 4-1 of this report, the maximum ΔRT_{NDT} values for the VCSNS Unit 1 is 102.0°F. Since the maximum ΔRT_{NDT} are projected to be above 100°F, but below 200°F, four (4) capsules are required to be pulled per Table 1 of ASTM E185-82. To date, five (5) capsules have been pulled and tested with the last tested capsule being Capsule Z. However, Capsule Z does not satisfy the GALL-SLR guidance for the capsule to obtain a fluence at least equal to the 80-year RV projected fluence. VCSNS Unit 1 currently has only one untested capsule (Capsule Y), which was removed at 17.71 EFPY with a fluence of 7.01 x 10^{19} n/cm^2 and placed in storage in the spent fuel pool.

It is recommended that, in order to satisfy the GALL-SLR guidance for the 80-year operating license for a capsule to be withdrawn and tested between one and two times the peak RV wall neutron fluence at the end of SPEO, Capsule Y should be reinserted into a 17° octagonally symmetric location (107°, 287°, or 343°) to be irradiated further. Capsule Y needs to be irradiated in a 17° location for a minimum of 4.9 EFPY[1] to experience the end of SPEO fluence of 9.06 x 10^{19} n/cm^2 (conservatively based on the fluence projection in Section 2 which includes a 10% bias on the periphery fuel assemblies), but it should be withdrawn before receiving 2 x the 80-year fluence, i.e., 1.73 x 10^{20} n/cm^2 (conservatively based on the fluence projection in Section 2 which excludes a 10% bias on the periphery fuel assemblies), at 22.4 EFPY. To assist in asset management and because Capsule Y is the last available capsule in the VCSNS RV surveillance program, it is recommended that a potential 100-year operating period also be considered. A projected 90 EFPY (90% capacity factor over 100 years) peak RV fluence of 1.14 x 10^{20} n/cm^2 (with 10% bias) should be achieved after a minimum of 10.5 EFPY of additional irradiation.

Conclusion

In order to meet all of the conditions described above, it is recommended that Capsule Y be exposed to a minimum of 10.5 EFPY of operation to experience the minimum 100-year fluence of 1.14×10^{20} n/cm^2. Assuming that Capsule Y is reinserted during Refueling Outage 30 in the Fall of 2027, prior to Cycle 31, this fluence will be achieved at a plant EFPY of 48.2 EFPY. Based on an average fuel cycle length of 1.33 EFPY/cycle, i.e., ~90% capacity factor, Capsule Y will need 8 cycles to achieve the additional 10.5 EFPY of irradiation (10.5 EFPY / 1.33 EFPY/cycle), which means the projected removal of Capsule Y following Cycle 38 (Fall of 2039). This means that Capsule Y will be reinserted at 37.7 EFPY at EOC 30 and removed at 48.4 EFPY at EOC 38 as this is the nearest outage to the suggested removal of 48.2 EFPY. These dates and cycle numbers are only estimates and may change as actual plant operation may differ from the assumptions used here.

1 The RV and capsule fluence projections are provided in Section 2 with and without a 10% bias on the periphery fuel assemblies. To ensure that the appropriate fluence is achieved, when the minimum capsule exposure is calculated, the RV fluence is based on the biased projection, while the capsule fluence is based on the unbiased fluence. When the maximum capsule fluence in calculated, the RV fluence is based on the unbiased projection, while the capsule fluence is based on the biased projection. This mixture of fluence data was conservatively applied to ensure the desired/minimum fluence was achieved and to prevent the maximum fluence from being exceeded.

Table 7-1 VCSNS Unit 1 Recommended Surveillance Capsule Withdrawal Schedule

Capsule	Location (deg)	Capsule Lead Factor	Removal Time[(a)] (EFPY)	Capsule Fluence (n/cm^2, E > 1.0 MeV)[(b)]
U	343	3.04	1.13 (EOC 1)	6.75×10^{18}
V	107	3.34	2.93 (EOC 3)	1.54×10^{19}
X	287	3.54	5.04 (EOC5)	2.51×10^{19}
W	110	3.21	11.21 (EOC 10)	4.63×10^{19}
Z	340	3.10	16.36 (EOC 14)	6.53×10^{19}
Y[(c)]	290	3.09	17.71 (EOC 15)	7.01×10^{19}
	107[(c)]	~3.5	48.2[(c)]	1.14×10^{20}

Notes:

(a) Effective full power years from plant startup. End of Cycle (EOC) value given in parenthesis. Note that core thermal power was uprated from 2775 to 2900 MW_{th} starting with operating Cycle 10.

(b) Fluence values were taken from Table 2-1.

(c) Capsule Y will be reinserted during Refueling Outage 30 in the Fall of 2027 (prior to Cycle 31) into location 107° (or symmetric locations 287° or 343°), which is projected to occur at 37.7 EFPY. Capsule Y will achieve the peak 100-year fluence, 1.14×10^{20} n/cm^2 (E > 1.0 MeV), before removal, which is calculated to require another 10.5 EFPY of operation (37.7 + 10.5 = 48.2).

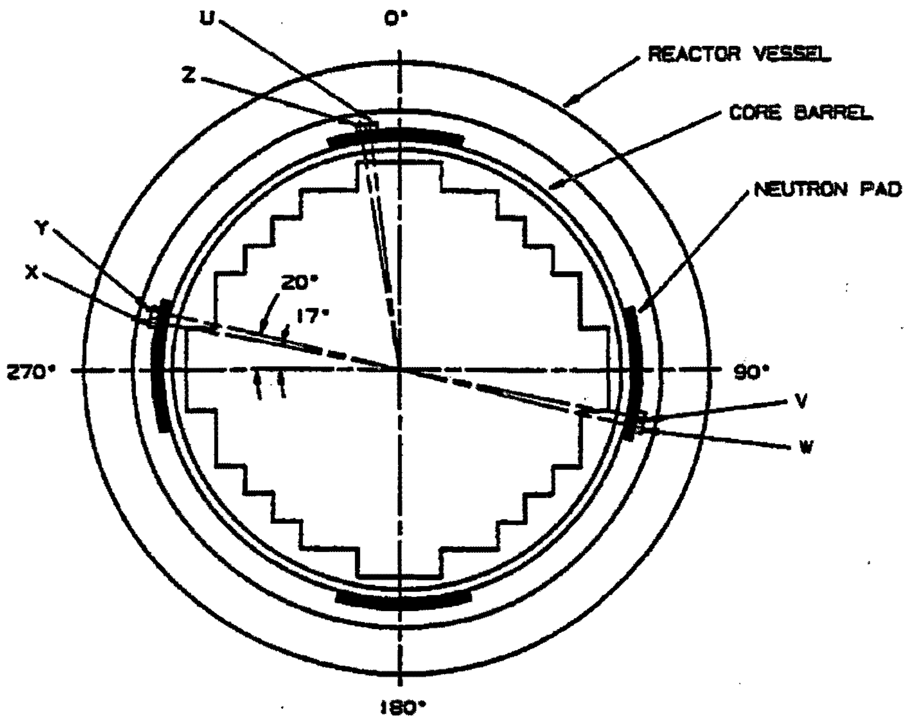

Figure 7-1 Original Arrangement of Surveillance Capsules in the VCSNS Unit 1 Reactor Vessel

8 REFERENCES

1. Code of Federal Regulations, 10 CFR Part 54.3, "Definitions," U.S. Nuclear Regulatory Commission, Federal Register, dated August 28, 2007.

2. U.S. Nuclear Regulatory Commission, Office of Nuclear Regulatory Research, Regulatory Guide 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence," March 2001. *[Agencywide Documents Access and Management System (ADAMS) Accession Number ML010890301]*

3. Code of Federal Regulations, 10 CFR 50, Appendix G, "Fracture Toughness Requirements," U.S. Nuclear Regulatory Commission, Federal Register, November 29, 2019.

4. Westinghouse Report WCAP-18124-NP-A, Revision 0, "Fluence Determination with RAPTOR-M3G and FERRET," July 2018.

5. Westinghouse Report WCAP-18709-NP, Revision 1, "V.C. Summer Nuclear Station Unit 1 Subsequent License Renewal: Reactor Pressure Vessel Extended Beltline Neutron Exposure Evaluation," February 2022.

6. Code of Federal Regulations, 10 CFR 50.61, "Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events," U.S. Nuclear Regulatory Commission, Federal Register, November 29, 2019.

7. Westinghouse PWROG Report PWROG-21037-NP, Revision 1, "Determination of Unirradiated RT_{NDT} and Upper-Shelf Energy Values of the V.C. Summer Unit 1 Reactor Vessel Materials," May 2022.

8. U.S. Nuclear Regulatory Commission, Office of Nuclear Regulatory Research, Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials," May 1988. *[ADAMS Accession Number ML003740284]*

9. Westinghouse PWROG Report PWROG-15109-NP-A, Revision 0, "PWR Pressure Vessel Nozzle Appendix G Evaluation," January 2020. [*ADAMS Accession Number ML20024E573*]

10. Westinghouse Report WCAP-16298-NP, Revision 0, "Analysis of Capsule Z from the South Carolina Electric & Gas Company V. C. Summer Reactor Vessel Radiation Surveillance Program," August 2004.

11. ASME Boiler and Pressure Vessel (B&PV) Code, Section III, Division 1, Subarticle NB-2300, "Fracture Toughness Requirements for Material and Section NB-3210, "Requirements for Acceptability," ASME International."

12. NUREG-0800, Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants, Chapter 5 of LWR Edition, Branch Technical Position 5-3, "Fracture Toughness Requirements," Revision 4, U.S. Nuclear Regulatory Commission, March 2019. *[ADAMS Accession Number ML18338A516]*

13. Appendix G to the 1998 through the 2000 Addenda Edition of the ASME Boiler and Pressure Vessel (B&PV) Code, Section XI, Division 1, "Fracture Toughness Criteria for Protection Against Failure."

14. Westinghouse Report WCAP-16305-NP, Revision 0, "V. C. Summer Heatup and Cooldown Limit Curves for Normal Operation," August 2004.

15. ASTM E185-82, "Standard Practice for Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels," American Society for Testing and Materials, 1982.

16. Code of Federal Regulations, 10 CFR 50, Appendix H, "Reactor Vessel Material Surveillance Program Requirements," U.S. Nuclear Regulatory Commission, Federal Register, October 2, 2020.

17. NUREG-1801, Revision 2, "Generic Aging Lessons Learned (GALL) Report," U.S. Nuclear Regulatory Commission, December 2010. *[ADAMS Accession Number ML103490041]*

18. NUREG-2191, Volume 2, "Generic Aging Lessons Learned for Subsequent License Renewal (GALL-SLR) Report," U.S. Nuclear Regulatory Commission, July 2017. *[ADAMS Accession Number ML17187A204]*

19. Westinghouse Report WCAP-9234, "South Carolina Electric and Gas Company Virgil C. Summer Nuclear Plant Unit No. 1 Reactor Vessel Radiation Surveillance Program," January 1978.

20. ASTM E185-73, "Standard Recommended Practice for Surveillance Tests for Nuclear Reactor Vessels," American Society for Testing and Materials, 1973.

21. Virgil C. Summer Nuclear Station Unit 1 Updated Final Safety Analysis Report (UFSAR), May 31, 2018.

22. NRC Regulatory Issue Summary 2014-11, "Information on Licensing Applications for Fracture Toughness Requirements for Ferritic Reactor Coolant Pressure Boundary Components," U.S. Nuclear Regulatory Commission, October 2014. *[ADAMS Accession No. ML14149A165]*

APPENDIX A CREDIBILITY EVALUATION OF THE VCSNS UNIT 1 SURVEILLANCE PROGRAM

Regulatory Guide 1.99, Revision 2 (Reference A-1) describes general procedures acceptable to the NRC staff for calculating the effects of neutron radiation embrittlement of the low-alloy steels currently used for light-water-cooled RVs. Positions 2.1 and 2.2 of Regulatory Guide 1.99, Revision 2, describe the method for calculating the adjusted reference temperature and Charpy upper-shelf energy of RV beltline materials using surveillance capsule data. The methods of Positions 2.1 and 2.2 can only be applied when two or more credible surveillance data sets become available from the reactor in question.

To date there have been five (5) surveillance capsules removed and tested from the VCSNS Unit 1 RV. In accordance with Regulatory Guide 1.99, Revision 2, the credibility of the surveillance data will be judged based on five criteria.

Table A-1 reviews the five criteria in Regulatory Guide 1.99, Revision 2. The following subsections evaluate each of these five criteria for VCSNS Unit 1 in order to determine the credibility of the surveillance data for use in neutron radiation embrittlement calculations.

Table A-1 Regulatory Guide 1.99, Revision 2, Credibility Criteria

Criterion No.	Description
1	Materials in the capsules should be those judged most likely to be controlling with regard to radiation embrittlement.
2	Scatter in the plots of Charpy energy versus temperature for the irradiated and unirradiated conditions should be small enough to permit the determination of the 30 ft-lbs temperature and upper-shelf energy unambiguously.
3	When there are two or more sets of surveillance data from one reactor, the scatter of ΔRT_{NDT} values about a best-fit line drawn as described in Regulatory Position 2.1 normally should be less than 28°F for welds and 17°F for base metal. Even if the fluence range is large (two or more orders of magnitude), the scatter should not exceed twice those values. Even if the data fail this criterion for use in shift calculations, they may be credible for determining decrease in upper-shelf energy if the upper shelf can be clearly determined, following the definition given in ASTM E185-82.
4	The irradiation temperature of the Charpy specimens in the capsule should match the vessel wall temperature at the cladding/base metal interface within +/- 25°F.
5	The surveillance data for the correlation monitor material in the capsule should fall within the scatter band of the database for that material.

A.1 VCSNS UNIT 1 CREDIBILITY EVALUATION

Criterion 1: Materials in the capsules should be those judged most likely to be controlling with regard to radiation embrittlement.

The VCSNS Unit 1 RV consists of the following beltline region materials, which likely would have been considered at the time the surveillance program was designed and licensed:

- Intermediate Shell 11-1 (Heat # A9154-1)
- Intermediate Shell 11-2 (Heat # A9153-2)
- Lower Shell 10-1 (Heat # C9923-2)
- Lower Shell 10-2 (Heat # C9923-1)
- Intermediate Shell Long. Weld Seams BC & BD (Heat # 4P4784, Flux Type Linde 124, Lot # 3930)
- Intermediate to Lower Shell Circ. Weld Seam AB (Heat # 4P4784, Flux Type Linde 124, Lot # 3930)
- Lower Shell Long. Weld Seams BA & BB (Heat # 4P4784, Flux Type Linde 124, Lot # 3930)

The VCSNS Unit 1 surveillance program utilizes longitudinal and transverse test specimens from intermediate shell plate A9154-1. The surveillance weld metal was fabricated with weld wire Heat # 4P4784, Linde 124 Flux, Lot # 3930.

At the time the VCSNS Unit 1 surveillance program material was selected it was believed that copper and phosphorus were the elements most important to embrittlement of the RV steels. The Intermediate Shell Plate A9154-1 had the highest Initial RT_{NDT} and one of the lowest initial USE of all plate materials in the beltline region. In addition, the Intermediate Shell Plate A9154-1 had the highest content of copper and phosphorus of all the other beltline plate materials. Therefore, based on the highest initial RT_{NDT}, the lowest initial USE and the highest copper and nickel content of all plate materials, Intermediate Shell Plate A9154-1 was chosen for the surveillance program.

The weld material in the VCSNS Unit 1 surveillance program was chosen as it is made of the same heat, flux type, and lot # as all of the RV beltline welds.

Based on the discussion, Criterion 1 is met for the VCSNS Unit 1 surveillance program.

Criterion 2: Scatter in the plots of Charpy energy versus temperature for the irradiated and unirradiated conditions should be small enough to permit the determination of the 30 ft-lbs temperature and upper-shelf energy unambiguously.

Based on engineering judgment, the scatter in the data presented in these plots, as documented in WCAP-16298-NP (Refence A-2), is small enough to permit the determination of the 30 ft-lb temperature and the upper-shelf energy of the VCSNS Unit 1 surveillance materials unambiguously.

<u>Hence, Criterion 2 is met for the VCSNS Unit 1 surveillance program.</u>

Criterion 3: When there are two or more sets of surveillance data from one reactor, the scatter of ΔRT_{NDT} values about a best-fit line drawn as described in Regulatory Position 2.1 normally should be less than 28°F for welds and 17°F for base metal. Even if the fluence range is large (two or more orders of magnitude), the scatter should not exceed twice those values. Even if the data fail this criterion for use in shift calculations, they may be credible for determining decrease in upper-shelf energy if the upper shelf can be clearly determined, following the definition given in ASTM E185-82.

The functional form of the least squares method as described in Regulatory Position 2.1 will be utilized to determine a best-fit line for this data and to determine if the scatter of these ΔRT_{NDT} values about this line is less than 28°F for welds and less than 17°F for the forgings.

Resulting is the calculation of the best-fit line as described in Regulatory Position 2.1 of Regulatory Guide 1.99, Revision 2. In addition, the recommended NRC methods for determining credibility will be followed. The NRC methods were presented to the industry at a meeting held by the NRC on February 12 and 13, 1998 (Reference A-3). At this meeting the NRC presented five cases. Of the five cases, Case 1 ("Surveillance data available from plant but no other source") most closely represents the situation for the VCSNS Unit 1 surveillance forging and weld material.

Case 1: Evaluation of VCSNS Unit 1 Data Only

Following the NRC Case 1 guidelines, the VCSNS Unit 1 surveillance plates and weld metal (Heat # 4P4784) will be evaluated using the VCSNS Unit 1 data. Table A-2 provides the calculation of the interim CFs. Only VCSNS Unit 1 data is being considered; therefore, no temperature adjustment or chemistry adjustments are required.

Table A-2 Calculation of Interim Chemistry Factors for the Credibility Evaluation Using VCSNS Unit 1 Surveillance Capsule Data Only

Material	Capsule	Capsule Fluence[(a)] (x 10^{19} n/cm^2, E > 1.0 MeV)	FF[(b)]	Measured ΔRT_{NDT}[(c)] (°F)	FF*ΔRT_{NDT} (°F)	FF^2
Intermediate Shell 11-1 (Longitudinal)	U	0.675	0.890	36.1	32.1	0.792
Intermediate Shell 11-1 (Longitudinal)	V	1.54	1.119	53.2	59.6	1.253
Intermediate Shell 11-1 (Longitudinal)	X	2.51	1.247	38.3	47.8	1.555
Intermediate Shell 11-1 (Longitudinal)	W	4.63	1.387	66.2	91.8	1.924
Intermediate Shell 11-1 (Longitudinal)	Z	6.53	1.451	98.9	143.5	2.106
Intermediate Shell 11-1 (Transverse)	U	0.675	0.890	14.5	12.9	0.792
Intermediate Shell 11-1 (Transverse)	V	1.54	1.119	32.1	35.9	1.253
Intermediate Shell 11-1 (Transverse)	X	2.51	1.247	26.7	33.3	1.555
Intermediate Shell 11-1 (Transverse)	W	4.63	1.387	57.8	80.2	1.924
Intermediate Shell 11-1 (Transverse)	Z	6.53	1.451	87.0	126.3	2.106
				SUM:	663.4	15.261
	$CF_{11\text{-}1} = \Sigma(FF * \Delta RT_{NDT}) \div \Sigma(FF^2) = (663.4) \div (15.261) = 43.5°F$					
Surveillance Weld	U	0.675	0.890	22.7	20.2	0.792
Surveillance Weld	V	1.54	1.119	47.0	52.6	1.253
Surveillance Weld	X	2.51	1.247	22.7	28.3	1.555
Surveillance Weld	W	4.63	1.387	43.5	60.3	1.924
Surveillance Weld	Z	6.53	1.451	65.2	94.6	2.106
				SUM:	256.1	7.630
	$CF_{Surv.\ Weld} = \Sigma(FF * \Delta RT_{NDT}) \div \Sigma(FF^2) = (256.1) \div (7.630) = 33.6°F$					

Notes:

(a) Fluence taken from Table 2-1.

(b) FF = fluence factor = $f^{(0.28 - 0.10 * \log(f))}$.

(c) Measured ΔRT_{NDT} taken from WCAP-16298-NP (Reference A-2).

The scatter of ΔRT_{NDT} values about the functional form of a best-fit line drawn as described in Regulatory Position 2.1 is presented in Table A-3.

Table A-3 VCSNS Unit 1 Surveillance Capsule Data Scatter about the Best-Fit Line

Material	Capsule	CF[(a)] (Slope$_{best-fit}$) (°F)	Capsule Fluence[(b)] (x 10^{19} n/cm^2, E > 1.0 MeV)	FF[(c)]	Measured ΔRT_{NDT}[(d)] (°F)	Predicted ΔRT_{NDT}[(e)] (°F)	Scatter ΔRT_{NDT}[(f)] (°F)	<17°F (Plate) <28°F (Weld)
Intermediate Shell 11-1 (Longitudinal)	U	43.5	0.675	0.890	36.1	38.7	2.6	Yes
	V	43.5	1.54	1.119	53.2	48.7	4.5	Yes
	X	43.5	2.51	1.247	38.3	54.2	15.9	Yes
	W	43.5	4.63	1.387	66.2	60.3	5.9	Yes
	Z	43.5	6.53	1.451	98.9	63.1	35.8	No
Intermediate Shell 11-1 (Transverse)	U	43.5	0.675	0.890	14.5	38.7	24.2	No
	V	43.5	1.54	1.119	32.1	48.7	16.6	Yes
	X	43.5	2.51	1.247	26.7	54.2	27.5	No
	W	43.5	4.63	1.387	57.8	60.3	2.5	Yes
	Z	43.5	6.53	1.451	87.0	63.1	23.9	No
Surveillance Weld (Heat # 4P4784)	U	33.6	0.675	0.890	22.7	29.9	7.2	Yes
	V	33.6	1.54	1.119	47.0	37.6	9.4	Yes
	X	33.6	2.51	1.247	22.7	41.9	19.2	Yes
	W	33.6	4.63	1.387	43.5	46.6	3.1	Yes
	Z	33.6	6.53	1.451	65.2	48.8	16.4	Yes

Notes:

(a) CF calculated in Table A-2 of this report.

(b) Fluence taken from Table 2-1 of this report.

(c) FF = fluence factor = $f^{(0.28 - 0.10*\log(f))}$.

(d) Measured ΔRT_{NDT} taken from WCAP-16298-NP (Reference A-2).

(e) Predicted ΔRT_{NDT} = CF x FF

(f) Scatter ΔRT_{NDT} = Absolute Value [Predicted ΔRT_{NDT} – Measured ΔRT_{NDT}].

The scatter of ΔRT_{NDT} values about the best-fit line, drawn as described in Regulatory Guide 1.99, Rev. 2, Position 2.1, should be less than 17°F for base metal and 28°F for welds. From a statistical point of view, +/- 1σ would be expected to encompass 68% of the data. Table A-3 indicates that the Intermediate Shell Plate 11-1 (Heat # A9154-1) has 6 of the 10 surveillance data points falling inside the +/- 1σ of 17°F scatter band for surveillance plate materials. Therefore, 60% of the data are bounded (6/10 x 100) and the surveillance plate data are deemed "non-credible" per the third criterion.

Table A-3 indicates that the Surveillance Weld has 5 of the 5 surveillance data points falling inside the +/- 1σ of 28°F scatter band for surveillance weld materials. Therefore, 100% of the data are bounded and the surveillance weld data are deemed "credible" per the third criterion.

Criterion 4: The irradiation temperature of the Charpy specimens in the capsule should match the vessel wall temperature at the cladding/base metal interface within +/- 25°F.

The capsule specimens are located in the reactor between the neutron shielding pads and the vessel wall and are positioned opposite the center of the core. The test capsules are in guide tubes attached to the neutron shielding pads. The location of the specimens with respect to the RV beltline provides assurance that the RV wall and the specimens experience equivalent operating conditions such that the temperatures will not differ by more than 25°F. Hence, this criterion is met.

Criterion 5: The surveillance data for the correlation monitor material in the capsule should fall within the scatter band of the database for that material.

The VCSNS Unit 1 surveillance program does not contain correlation monitor material. Hence, this criterion is not applicable to the VCSNS Unit 1 surveillance program.

CONCLUSION

Based on the preceding responses to the 5 criteria of Regulatory Guide 1.99, Revision 2, Section B, the VCSNS Unit 1 surveillance data for the Intermediate Shell Plate 11-1 (Heat # A9154-1) is deemed non-credible and the Surveillance Weld (Heat # 4P4784) is deemed credible.

A.2 REFERENCES

A-1. U.S. Nuclear Regulatory Commission, Office of Nuclear Regulatory Research, Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials," May 1988. *[ADAMS Accession Number ML003740284]*

A-2. Westinghouse Report, WCAP-16298-NP, Revision 0, "Analysis of Capsule Z from the South Carolina Electric & Gas Company V. C. Summer Reactor Vessel Radiation Surveillance Program," August 2004.

A-3. K. Wichman, M. Mitchell, and A. Hiser, US NRC, "Generic Letter 92-01 and RPV Integrity Assessment, Status, Schedule, and Issues," NRC/Industry Workshop on RPV Integrity Issues, February 12, 1998. *[ADAMS Accession Number ML110070570]*

APPENDIX B EMERGENCY RESPONSE GUIDELINES

The Emergency Response Guideline (ERG) limits were developed to establish guidance for operator action in the event of an emergency situation, such as a PTS event (Reference B-1). Generic categories of limits were developed for the guidelines based on the limiting inside surface RT_{NDT}. These generic categories were conservatively generated for the Westinghouse Owners Group (WOG) to be applicable to all Westinghouse plants.

The highest value of RT_{NDT} for which the generic category ERG limits were developed is 250°F for a longitudinal flaw and 300°F for a circumferential flaw. Therefore, if the limiting vessel material has an RT_{NDT} that exceeds 250°F for a longitudinal flaw or 300°F for a circumferential flaw, plant-specific ERG P-T limits must be developed.

The ERG category is determined by the magnitude of the limiting RT_{NDT} value, which is calculated the same way as the RT_{PTS} values are calculated in Section 4 of this report. The material with the highest RT_{NDT} defines the limiting material, which for VCSNS Unit 1 is the Intermediate Shell 11-1. Table B-1 identifies ERG category limits and the limiting material RT_{NDT} values at 72 EFPY for VCSNS Unit 1.

Table B-1 Evaluation of VCSNS Unit 1 ERG Limit Category

ERG Pressure-Temperature Limits (Reference B-1)	
Applicable RT_{NDT} Value[(a)]	**ERG P-T Limit Category**
RT_{NDT} < 200°F	Category I
200°F < RT_{NDT} < 250°F	Category II
250°F < RT_{NDT} < 300°F	Category IIIb
Limiting RT_{NDT} Value[(b)]	
Limiting Reactor Vessel Material	**RT_{NDT} Value @ 72 EFPY**
Intermediate Shell 11-1	152.5

Note(s):

(a) Longitudinally oriented flaws are applicable only up to 250°F; circumferentially oriented flaws are applicable up to 300°F.

(b) Limiting value taken from Table 4-1.

Per the ERG limit guidance document (Reference B-1), some vessels do not change categories for operation through the end of license. However, when a vessel does change ERG categories between the beginning and end of operation, a plant-specific assessment must be performed to determine at what operating time the category changes. Thus, the ERG classification need not be changed until the operating cycle during which the maximum vessel value of actual or estimated real-time RT_{NDT} exceeds the limit on its current ERG category.

Conclusion of ERG P-T Limit Categorization

The limiting VCSNS Unit 1 RV material RT_{NDT} values do not exceed 200°F. Therefore, VCSNS Unit 1 will remain in Category I through SPEO.

B.1 REFERENCES

B-1. Westinghouse Owners Group Document, HF04BG, "Background Information for Westinghouse Owners Group Emergency Response Guidelines, Critical Safety Function Status Tree, F-0.4 Integrity, HP/LP-Rev. 3," March 31, 2014.

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Enclosure 4
Attachment 4

WCAP-18754-NP, REVISION 2

Virgil C. Summer (VCSNS) Unit 1
Dominion Energy South Carolina, Inc. (DESC)

WCAP-18754-NP
Revision 2

August 2022

V.C. Summer Nuclear Station Unit 1 Subsequent License Renewal: Reactor Pressure Vessel Internals Neutron Exposure Evaluation

Westinghouse

WCAP-18754-NP
Revision 2

V.C. Summer Nuclear Station Unit 1 Subsequent License Renewal: Reactor Pressure Vessel Internals Neutron Exposure Evaluation

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RECORD OF REVISIONS

Revision	Description of Changes
0	Initial issue.
1	Changes made to resolve customer comments attached to archive record. The global editorial change from “$<10^{20}$” to “$< 10^{20}$” was not tracked.
2	Resolved a corrective action to correct the calculated exposure region for several components.

TABLE OF CONTENTS

LIST OF TABLES v

LIST OF FIGURES vi

1 INTRODUCTION 1
2 NEUTRON TRANSPORT ANALYSIS 1
3 REACTOR INTERNALS 4
4 EXPOSURE RESULTS 5
5 REFERENCES 19

LIST OF TABLES

Table 4-1. Maximum Fast Neutron (E > 1.0 MeV) Fluence, Iron Atom Displacements and Exposure Region Assignments at 72 EFPY 7

LIST OF FIGURES

Figure 2-1. A View of the Reactor Geometry Clipped at the Core Midplane 2
Figure 2-2. Oblique View of the Reactor Geometry 3
Figure 3-1. Excerpt from MRP-191 Revision 2 Showing Typical Westinghouse Reactor Internals 4
Figure 4-1. Cross-Sectional Views of the Reactor Internals at Selected Azimuthal Locations 15
Figure 4-2. Regional Map of the Core Baffle 16
Figure 4-3. Regional Map of the Formers 17
Figure 4-4. Regional Maps at Select Axial Cross-Sections of the Reactor Internals 18

1 INTRODUCTION

This report describes a comparison between the calculated fast neutron (E > 1.0 MeV) exposure received by reactor internals components in V.C. Summer Nuclear Station (VCSNS) Unit 1 (after 80 years of plant operation (72 effective full power years (EFPY)) against the estimated component fluence for a Westinghouse plant after 80 years of operation as documented in MRP-191 Revision 2, "Materials Reliability Program: Screening, Categorization, and Ranking of Reactor Internals Components for Westinghouse and Combustion Engineering PWR Design (MRP-191, Reference 1)." Revision 2 of MRP-191 documents the process for and updates the categorization results in support of developing the basis for aging management programs in subsequent license renewals.

The neutron transport methodology used to generate the data provided in this attachment was developed consistent with the guidance of Regulatory Guide 1.190 (Reference 2). It is also consistent with the methodology described in WCAP-18124-NP-A (Reference 3) that was approved by the United States Nuclear Regulatory Commission (USNRC). The methodology described in Reference 3 was approved for calculating exposures of the reactor pressure vessel beltline. Presently, there are no regulator-approved methods for calculating exposures of reactor internals components.

2 NEUTRON TRANSPORT ANALYSIS

The neutron transport analyses utilized for evaluation of the reactor internals are described in WCAP-18709-NP (Reference 4). To summarize the analysis method, discrete ordinates transport calculations were performed on a fuel-cycle-specific basis to determine the neutron and gamma ray environment within the reactor geometry. In those analyses, anisotropic scattering was treated with a P_3 Legendre expansion. The angular discretization was modeled with an S_{16} order of angular quadrature. Material cross-section data were based on data derived from the ENDF/B-VI cross section database. Energy-dependent and space-dependent core power distributions, as well as system operating temperatures, were treated on a fuel-cycle-specific basis. The uncertainties associated with the method of analysis apply equally to this analysis of the reactor internals.

The discrete ordinates transport calculations described in WCAP-18709-NP yielded a three-dimensional database comprised of fuel-cycle specific scalar fluence rate data, which was interrogated to generate the projected component-specific exposure data presented herein for VCSNS Unit 1 after 72 EFPY, or 80 years of operation. For those projections, a 10% positive bias was applied to the relative powers of the fuel assemblies on the periphery of the core and on the re-entrant corners of the core baffle.

The VCSNS Unit 1 reactor is a Westinghouse-designed, 3-loop pressurized water reactor with neutron panels (as opposed to thermal shields). The model geometry used in the transport analysis is described in WCAP-18709-NP. A view of the model geometry clipped at the core midplane is shown in Figure 2-1. Regarding the reactor internals, this figure reflects a single quadrant arrangement of the core, core baffle, core barrel, neutron panels and surveillance capsule holder with the carbon steel capsule contents. The RPV and other components and structures not making up the reactor internals are also shown. An oblique view of the model geometry is shown in Figure 2-2. Figure 2-2 also depicts the upper and lower core plates and the core former plates, which were explicitly included in the model, as well as the upper and lower reactor internals, which were modeled as a combination of discrete components and homogenized regions.

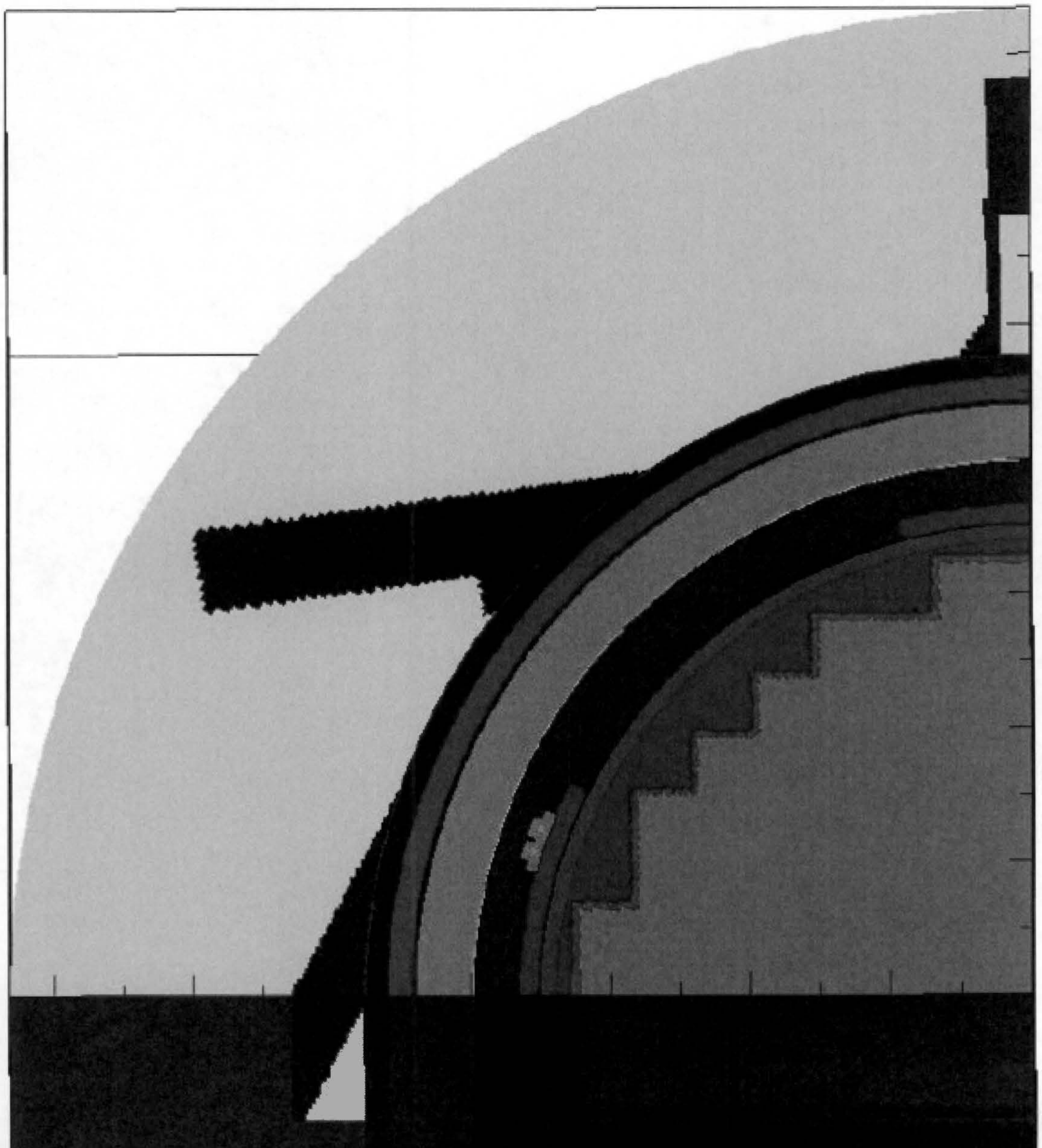

Figure 2-1. A View of the Reactor Geometry Clipped at the Core Midplane

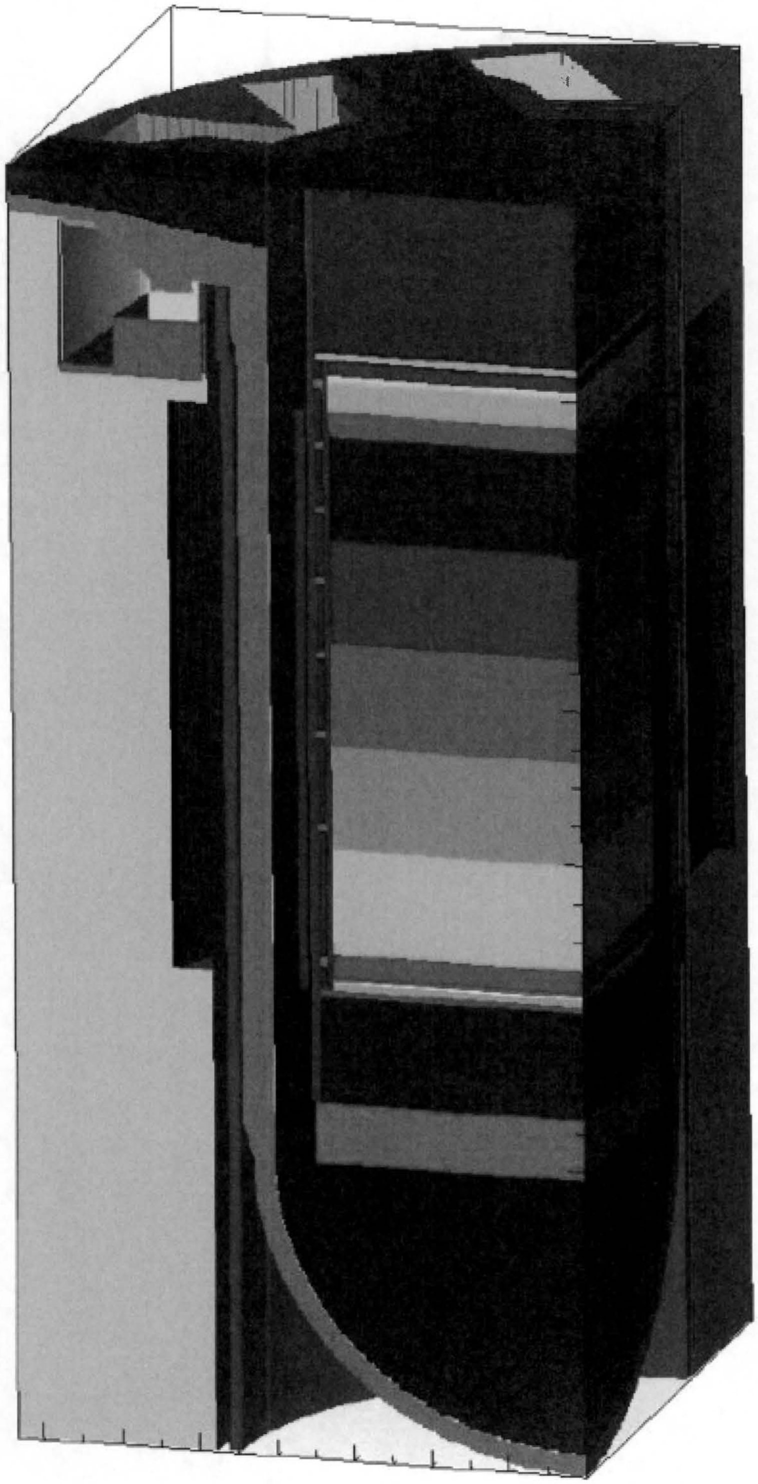

Figure 2-2. Oblique View of the Reactor Geometry

3 REACTOR INTERNALS

The excerpt from MRP-191 Revision 2 in Figure 3-1 shows the relative location of the component groups identified in the results in Section 4. The exposure analysis for the reactor internals components employs nominal design dimensions for the explicitly modeled components, such as the upper core plate, lower core plate, core baffle, formers and core barrel. The smaller internals subcomponents, such as fuel pins, inserts, locking caps, etc., were not explicitly modeled. Exposure of these smaller subcomponents was conservatively assigned based on the maximum fluence observed in the general area in which the components reside.

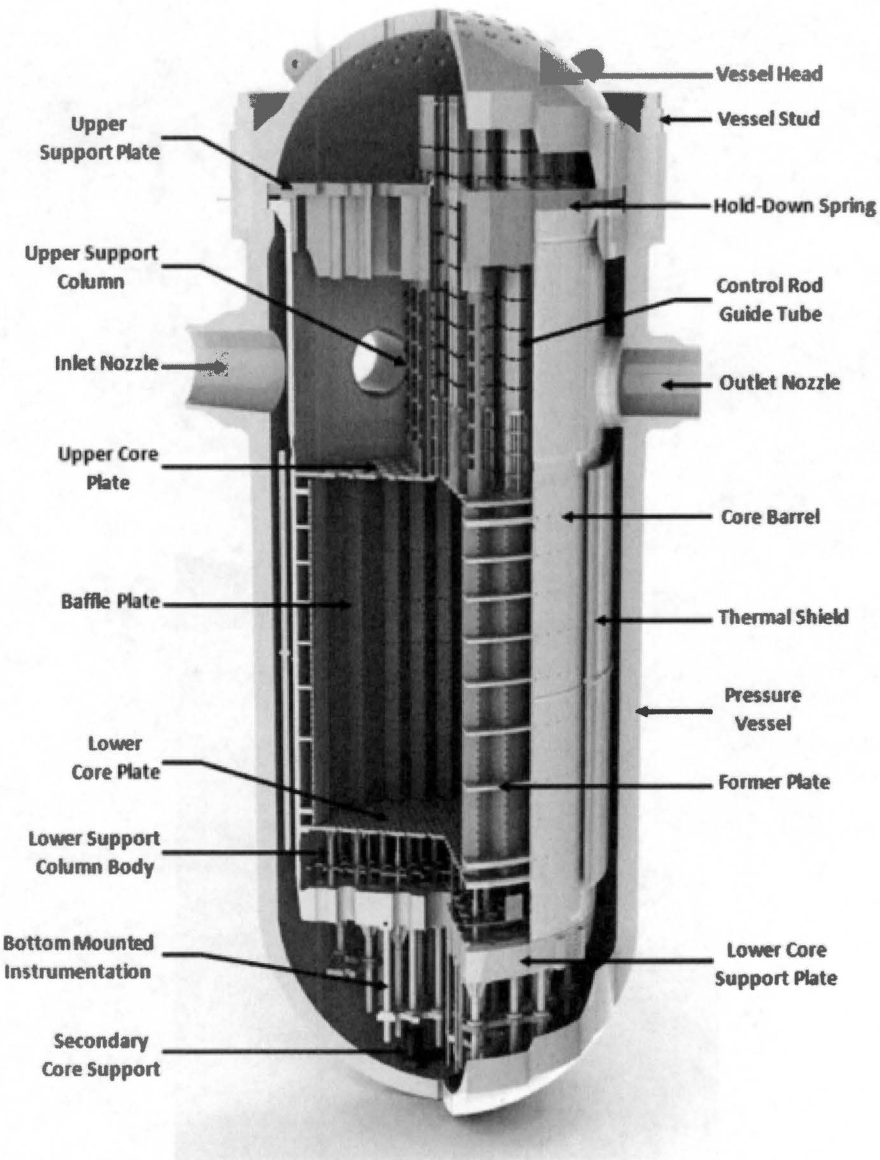

Figure 3-1. Excerpt from MRP-191 Revision 2 Showing Typical Westinghouse Reactor Internals

4 EXPOSURE RESULTS

Revision 2 of MRP-191 identifies radiation exposure criteria to be used in screening evaluations for reactor internals components. The exposure bin ranges as defined in Section 4.2.3 of MRP-191 Revision 2 were as follows:

- **Region 1**: $\phi t < 1x10^{20}$ n/cm^2
- **Region 2**: $1x10^{20}$ n/cm^2 (0.15 dpa) $\leq \phi t < 7x10^{20}$ n/cm^2
- **Region 3**: $7x10^{20}$ n/cm^2 (1 dpa) $\leq \phi t < 1x10^{21}$ n/cm^2
- **Region 4**: $1x10^{21}$ n/cm^2 (1.5 dpa) $\leq \phi t < 1x10^{22}$ n/cm^2
- **Region 5**: $1x10^{22}$ n/cm^2 (15 dpa) $\leq \phi t < 5x10^{22}$ n/cm^2
- **Region 6**: $5x10^{22}$ n/cm^2 (75 dpa) $\leq \phi t$

where ϕt (fluence) is for neutron energies with E > 1.0 MeV.

The calculated exposure region for each component in Table 4-1 was calculated by applying the three-dimensional fluence rates developed from the transport analyses described in WCAP-18709-NP (Reference 4) and then assigning each component to the appropriate exposure "bin" previously described.

The calculated maximum fast neutron (E > 1.0 MeV) fluence, iron atom displacements, and the calculated exposure region, as well as the estimated exposure region as defined in Section 4.2.3 of MRP-191 Revision 2 for each applicable component of the VCSNS Unit 1 reactor internals are reflected in Table 4-1. For the calculated neutron fluence and iron atom displacements in Table 4-1, the values reflect the neutron fluence at the specific location of the component, except for components for which the calculated fluence is identified by the range "$< 10^{20}$." An elevation was selected above the upper core plate and below the lower core plate beyond which the neutron fluence would be below the MRP-191 Region 1 level for all components. The components whose fluence is identified by this range are located either above or below the relevant cut off elevation. The calculated exposure region assigned in Table 4-1 could be based on the calculated fluence or the calculated iron atom displacements. That is, an item with fluence below the upper limit of Region 4 that has iron atom displacements greater than the lower limit of Region 5 would be assigned to Region 5. Conventional rounding methods are applied. The calculated fast neutron fluence, calculated iron atom displacements and calculated exposure region can be compared to the estimated exposure region and estimated fluence range in Table 4-7 of MRP-191 Revision 2, which are also reflected in the results.

Several plots were generated showing the distribution of the MRP-191 exposure regions across reactor internals components. Figure 4-1 shows a distribution of the fast neutron fluence at the azimuth of maximum fluence (0°), at 17° (the location of the surveillance capsule with the highest fluence rate), and at 45° (the azimuth of lowest fluence). Figure 4-2 shows a distribution of the fast neutron fluence on the baffle plates, which were most highly exposed on the reentrant corners. Figure 4-3 shows a distribution of the fast neutron fluence on the former plates, which were also most highly exposed on the reentrant corners. Figure 4-4 depicts the distribution of the fast neutron fluence on the core-facing side of the upper core plate and the lower core plate. These figures provide a visualization tool should evaluation of a component not explicitly described in Table 4-1 become necessary or if greater refinement is desired.

With the exception of the components below, the neutron exposures received by the reactor internal components were bounded by the estimated exposures in MRP-191 Revision 2. The following components were projected to receive neutron exposures higher than estimated by MRP-191 Revision 2 at 72 EFPY:

- Irradiation specimen guide bolts:

  The irradiation specimen guide bolts were estimated to be in Exposure Region 2 at 72 EFPY. The calculated fast neutron fluence at the radius and axial range of the irradiation specimen guide bolts was $2x10^{21}$ n/cm^2 with 2.6 displacements per iron atom (dpa), which put these bolts in Exposure Region 4.

- Irradiation specimen guides:

  The irradiation specimen guides were estimated to be in Exposure Region 2 at 72 EFPY. The calculated fast neutron fluence at the irradiation specimen guides was $6x10^{20}$ n/cm^2 (Exposure Region 2 fluence), but the iron atom displacements totaled more than 1 dpa, putting these components in Exposure Region 3.

- Irradiation specimen guide lock caps:

  The irradiation specimen guide lock caps were estimated to be in Exposure Region 2 at 72 EFPY. The calculated fast neutron fluence at the irradiation specimen guide lock caps was $6x10^{20}$ n/cm^2 (Exposure Region 2 fluence), but the iron atom displacements totaled more than 1 dpa, putting these components in Exposure Region 3.

- Neutron panel bolts:

  The neutron panel bolts were estimated to be in Exposure Region 4 at 72 EFPY. The calculated fast neutron fluence at the neutron panel bolts was $9x10^{21}$ n/cm^2 (Exposure Region 4 fluence), but the iron atom displacements totaled more than 15 dpa, putting these components in Exposure Region 5.

- Neutron panel locking devices:

  The neutron panel locking devices were estimated to be in Exposure Region 2 at 72 EFPY. The calculated fast neutron exposure at the neutron panel locking devices was $1x10^{21}$ n/cm^2 with 2.1 dpa, putting these components in Exposure Region 4.

For the irradiation specimen guide bolts, the fluence reported is that which occurs at the embedded end of the bolt near the inner surface of the neutron pad. For the neutron panel bolts, the fluence reported is that which occurs at the embedded end of the bolt near the inner surface of the core barrel.

The screening described in MRP-191 Revision 2 provides criteria upon which to gauge the susceptibility of reactor internals components to postulated aging-related degradation mechanisms (ARDM). For the reactor internals components determined herein to receive calculated neutron exposure less than or equal to the estimated exposure regions documented in MRP-191 Revision 2, the credible irradiation-related ARDMs are identified in the screening results in Section 5.2 of MRP-191 Revision 2. The calculated fast neutron ($E > 1.0$ MeV) fluence or iron atom displacements at 72 EFPY for the previously itemized components was higher than the exposure regions documented in MRP-191 Revision 2, potentially resulting in additional irradiation-related ARDMs screening in for these components. Further evaluation of these components is required to show continued applicability of the failure modes, effects, and criticality analysis results of MRP-191, Revision 2 and to apply the downstream results in MRP-227, Revision 2. Such an evaluation is outside of the scope of this analysis.

Table 4-1. Maximum Fast Neutron (E > 1.0 MeV) Fluence, Iron Atom Displacements and Exposure Region Assignments at 72 EFPY

Component Group	Component Description	Calculated			MRP-191 Estimated	
		Fluence (n/cm^2)	Iron Atom Displacements (dpa)	Region	Region	Fluence Range (n/cm^2)
Control Rod Guide Tube Assemblies and Flow Downcomers	Anti-rotation studs and nuts	$< 10^{20}$	< 0.15	1	1	$< 10^{20}$
	Bolts	$< 10^{20}$	< 0.15	1	1	$< 10^{20}$
	C-tubes	$6x10^{20}$	0.81	2	3	$7x10^{20}$ to $1x10^{21}$
	Enclosure pins	$< 10^{20}$	< 0.15	1	1	$< 10^{20}$
	Guide tube enclosures	$6x10^{20}$	0.81	2	3	$7x10^{20}$ to $1x10^{21}$
	Flanges, intermediate	$< 10^{20}$	< 0.15	1	1	$< 10^{20}$
	Flanges, lower	$6x10^{20}$	0.81	2	4	$1x10^{21}$ to $1x10^{22}$
	Guide plates/cards	$< 10^{20}$	< 0.15	1	2	$1x10^{20}$ to $7x10^{20}$
	Guide tube support pins	$1x10^{21}$	1.6	4	4	$1x10^{21}$ to $1x10^{22}$
	Housing plates	$< 10^{20}$	< 0.15	1	1	$< 10^{20}$
	Lock bars	$< 10^{20}$	< 0.15	1	1	$< 10^{20}$
	Sheaths	$6x10^{20}$	0.81	2	3	$7x10^{20}$ to $1x10^{21}$
	Cover plates	$< 10^{20}$	< 0.15	1	1	$< 10^{20}$
	Cover plate cap screws	$< 10^{20}$	< 0.15	1	1	$< 10^{20}$
	Cover plate locking caps and tie straps	$< 10^{20}$	< 0.15	1	1	$< 10^{20}$
	Support pin nuts	$6x10^{20}$	0.81	2	4	$1x10^{21}$ to $1x10^{22}$
	Water flow slot ligaments	$1x10^{20}$	0.21	2	3	$7x10^{20}$ to $1x10^{21}$
Mixing Devices	Mixing Devices	$1x10^{21}$	1.6	4	4	$1x10^{21}$ to $1x10^{22}$

Table 4-1. Maximum Fast Neutron (E > 1.0 MeV) Fluence, Iron Atom Displacements and Exposure Region Assignments at 72 EFPY

Component Group	Component Description	Calculated			MRP-191 Estimated	
		Fluence (n/cm^2)	Iron Atom Displacements (dpa)	Region	Region	Fluence Range (n/cm^2)
Upper Core Plate (UCP) and Fuel Alignment Pins	Fuel alignment pins	$3x10^{21}$	4.0	4	4	$1x10^{21}$ to $1x10^{22}$
	UCP	$1x10^{21}$	1.6	4	4	$1x10^{21}$ to $1x10^{22}$
	UCP insert	$3x10^{20}$	0.48	2	3	$7x10^{20}$ to $1x10^{21}$
	UCP insert bolts	$3x10^{20}$	0.48	2	3	$7x10^{20}$ to $1x10^{21}$
	UCP insert locking devices and dowel pins	$3x10^{20}$	0.48	2	3	$7x10^{20}$ to $1x10^{21}$
Upper Instrumentation Conduit and Support	Bolting	$< 10^{20}$	< 0.15	1	1	$< 10^{20}$
	Brackets, clamps terminal blocks, and conduit straps	$< 10^{20}$	< 0.15	1	3	$7x10^{20}$ to $1x10^{21}$
	Conduit seal assembly: body, tubesheets, tubesheet welds	$< 10^{20}$	< 0.15	1	1	$< 10^{20}$
	Conduit seal assembly: tubes	$< 10^{20}$	< 0.15	1	1	$< 10^{20}$
	Conduits	$< 10^{20}$	< 0.15	1	2	$1x10^{20}$ to $7x10^{20}$
	Flange base	$< 10^{20}$	< 0.15	1	1	$< 10^{20}$
	Locking Caps	$< 10^{20}$	< 0.15	1	1	$< 10^{20}$
	Support tubes	$< 10^{20}$	< 0.15	1	1	$< 10^{20}$

Table 4-1. Maximum Fast Neutron (E > 1.0 MeV) Fluence, Iron Atom Displacements and Exposure Region Assignments at 72 EFPY

Component Group	Component Description	Calculated			MRP-191 Estimated	
		Fluence (n/cm^2)	Iron Atom Displacements (dpa)	Region	Region	Fluence Range (n/cm^2)
Upper Support Column Assemblies	Adapters	$< 10^{20}$	< 0.15	1	1	$< 10^{20}$
	Bolts	$1x10^{21}$	1.6	4	4	$1x10^{21}$ to $1x10^{22}$
	Column bases	$1x10^{21}$	1.6	4	4	$1x10^{21}$ to $1x10^{22}$
	Column bodies	$6x10^{20}$	0.81	2	2	$1x10^{20}$ to $7x10^{20}$
	Extension tubes	$< 10^{20}$	< 0.15	1	1	$< 10^{20}$
	Flanges	$< 10^{20}$	< 0.15	1	1	$< 10^{20}$
	Lock keys	$< 10^{20}$	< 0.15	1	3	$7x10^{20}$ to $1x10^{21}$
	Lock caps[1]	$6x10^{20}$	0.81	2	--	--
	Nuts	$< 10^{20}$	< 0.15	1	1	$< 10^{20}$
Upper Support Plate Assembly – Flat Plate Design	Upper support plate	$< 10^{20}$	< 0.15	1	1	$< 10^{20}$
	Upper support ring or skirt	$< 10^{20}$	< 0.15	1	1	$< 10^{20}$
	Deep beam ribs	$< 10^{20}$	< 0.15	1	1	$< 10^{20}$
	Deep beam stiffeners	$< 10^{20}$	< 0.15	1	1	$< 10^{20}$
	Locking device	$< 10^{20}$	< 0.15	1	1	$< 10^{20}$

[1] Upper support column bases use locking caps instead of locking keys. Though not explicitly listed in MRP-191, the locking caps are similar in function and location to locking keys and are thus included. The MRP-191 inspection program is considered applicable because the fast neutron fluence at the lock caps is within the exposure range estimated for the lock keys.

Table 4-1. Maximum Fast Neutron (E > 1.0 MeV) Fluence, Iron Atom Displacements and Exposure Region Assignments at 72 EFPY

Component Group	Component Description	Calculated			MRP-191 Estimated	
		Fluence (n/cm^2)	Iron Atom Displacements (dpa)	Region	Region	Fluence Range (n/cm^2)
Baffle and Former Assembly	Baffle bolting lock devices	$1x10^{23}$	176	6	6	$\geq 5x10^{22}$
	Baffle-edge bolts	$1x10^{23}$	176	6	6	$\geq 5x10^{22}$
	Baffle plates	$1x10^{23}$	176	6	6	$\geq 5x10^{22}$
	Baffle-former bolts	$1x10^{23}$	176	6	6	$\geq 5x10^{22}$
	Barrel-former bolts	$1x10^{22}$	15	5	5	$1x10^{22}$ to $5x10^{22}$
	Former plates	$8x10^{22}$	114	6	6	$\geq 5x10^{22}$
Bottom-Mounted Instrumentation (BMI) Column Assemblies	BMI column bodies	$< 10^{20}$	< 0.15	1	1	$< 10^{20}$
	BMI column bolts	$< 10^{20}$	< 0.15	1	5	$1x10^{22}$ to $5x10^{22}$
	BMI column collars	$1x10^{22}$	19	5	5	$1x10^{22}$ to $5x10^{22}$
	BMI column cruciforms	$< 10^{20}$	< 0.15	1	5	$1x10^{22}$ to $5x10^{22}$
	BMI column extension bars	$2x10^{22}$	32	5	5	$1x10^{22}$ to $5x10^{22}$
	BMI column extension tubes	$< 10^{20}$	< 0.15	1	1	$< 10^{20}$
	BMI column locking devices	$1x10^{22}$	19	5	5	$1x10^{22}$ to $5x10^{22}$
	BMI column nuts	$2x10^{22}$	32	5	5	$1x10^{22}$ to $5x10^{22}$

Table 4-1. Maximum Fast Neutron (E > 1.0 MeV) Fluence, Iron Atom Displacements and Exposure Region Assignments at 72 EFPY

Component Group	Component Description	Calculated			MRP-191 Estimated	
		Fluence (n/cm^2)	Iron Atom Displacements (dpa)	Region	Region	Fluence Range (n/cm^2)
Core Barrel	Core barrel flange	$< 10^{20}$	< 0.15	1	1	$< 10^{20}$
	Upflow conversion core barrel plug - body	$1x10^{21}$	1.6	4	4	$1x10^{21}$ to $1x10^{22}$
	Upflow conversion core barrel plug - mandrel	$1x10^{21}$	1.6	4	4	$1x10^{21}$ to $1x10^{22}$
	Core barrel outlet nozzles	$< 10^{20}$	< 0.15	1	1	$< 10^{20}$
	Lower core barrel axial welds (MAW and LAW)	$1x10^{21}$	2.2	4	5	$1x10^{22}$ to $5x10^{22}$
	Lower core barrel girth welds (LGW and LFW)	$9x10^{21}$	14	4	5	$1x10^{22}$ to $5x10^{22}$
	Upper core barrel axial welds (UAW)	$< 10^{20}$	< 0.15	1	2	$1x10^{20}$ to $7x10^{20}$
	Upper core barrel girth welds (UFW and UGW)	$< 10^{20}$	< 0.15	1	2	$1x10^{20}$ to $7x10^{20}$
Flux Thimbles	Flux thimble tube plugs	$\geq 5x10^{22}$	> 75	6	6	$\geq 5x10^{22}$
	Flux thimbles (tubes)	$\geq 5x10^{22}$	> 75	6	6	$\geq 5x10^{22}$
Head Cooling Spray Nozzles	Head cooling spray nozzles	$< 10^{20}$	< 0.15	1	1	$< 10^{20}$

Table 4-1. Maximum Fast Neutron (E > 1.0 MeV) Fluence, Iron Atom Displacements and Exposure Region Assignments at 72 EFPY

Component Group	Component Description		Calculated			MRP-191 Estimated	
			Fluence (n/cm^2)	Iron Atom Displacements (dpa)	Region	Region	Fluence Range (n/cm^2)
Irradiation Specimen Guides	Irradiation specimen guides		$6x10^{20}$	**1.1**	**3**	2	$1x10^{20}$ to $7x10^{20}$
	Irradiation specimen guide bolts		**$2x10^{21}$**	**2.6**	**4**	2	$1x10^{20}$ to $7x10^{20}$
	Irradiation specimen guide lock caps		$6x10^{20}$	**1.1**	**3**	2	$1x10^{20}$ to $7x10^{20}$
	Irradiation specimen plug	Spring	$< 10^{20}$	< 0.15	1	1	$< 10^{20}$
		Dowel pin	$< 10^{20}$	< 0.15	1	1	$< 10^{20}$
		Plug	$< 10^{20}$	< 0.15	1	1	$< 10^{20}$
Lower Core Plate (LCP) and Fuel Alignment Pins	Fuel alignment pins		$3x10^{22}$	42	5	6	$\geq 5x10^{22}$
	LCP and manway bolts		$2x10^{22}$	32	5	5	$1x10^{22}$ to $5x10^{22}$
	LCP and manway locking devices		$2x10^{22}$	32	5	5	$1x10^{22}$ to $5x10^{22}$
	Lower core plate		$1x10^{22}$	19	5	5	$1x10^{22}$ to $5x10^{22}$
Lower Support Column Assemblies	Lower support column bodies		$6x10^{21}$	8.9	4	5	$1x10^{22}$ to $5x10^{22}$
	Lower support column bolts		$2x10^{22}$	32	5	5	$1x10^{22}$ to $5x10^{22}$
	Lower support column bolt locking devices		$2x10^{22}$	32	5	5	$1x10^{22}$ to $5x10^{22}$
	Lower support column nuts		$< 10^{20}$	< 0.15	1	5	$1x10^{22}$ to $5x10^{22}$
	Lower support column sleeves		$< 10^{20}$	< 0.15	1	2	$1x10^{20}$ to $7x10^{20}$

Table 4-1. Maximum Fast Neutron (E > 1.0 MeV) Fluence, Iron Atom Displacements and Exposure Region Assignments at 72 EFPY

Component Group	Component Description	Calculated			MRP-191 Estimated	
		Fluence (n/cm^2)	Iron Atom Displacements (dpa)	Region	Region	Fluence Range (n/cm^2)
Lower Support Casting or Forging	Lower support forging	$< 10^{20}$	< 0.15	1	1	$< 10^{20}$
Neutron Panels	Neutron panel bolts	**$9x10^{21}$**	**15**	**5**	4	$1x10^{21}$ to $1x10^{22}$
	Neutron panel locking devices	**$1x10^{21}$**	**2.1**	**4**	2	$1x10^{20}$ to $7x10^{20}$
	Neutron panels	$4x10^{21}$	6.5	4	4	$1x10^{21}$ to $1x10^{22}$
Radial Support Keys	Radial support key bolts	$< 10^{20}$	< 0.15	1	1	$< 10^{20}$
	Radial support key lock keys	$< 10^{20}$	< 0.15	1	1	$< 10^{20}$
	Radial support key dowels	$< 10^{20}$	< 0.15	1	1	$< 10^{20}$
	Radial support keys	$< 10^{20}$	< 0.15	1	1	$< 10^{20}$
Secondary Core Support (SCS) Assembly	SCS base plates	$< 10^{20}$	< 0.15	1	1	$< 10^{20}$
	SCS bolts	$< 10^{20}$	< 0.15	1	1	$< 10^{20}$
	SCS energy absorber	$< 10^{20}$	< 0.15	1	1	$< 10^{20}$
	SCS guide post	$< 10^{20}$	< 0.15	1	1	$< 10^{20}$
	SCS housing	$< 10^{20}$	< 0.15	1	1	$< 10^{20}$
	SCS lock keys	$< 10^{20}$	< 0.15	1	1	$< 10^{20}$
	Upper and lower tie plates	$< 10^{20}$	< 0.15	1	1	$< 10^{20}$

Table 4-1. Maximum Fast Neutron (E > 1.0 MeV) Fluence, Iron Atom Displacements and Exposure Region Assignments at 72 EFPY

Component Group	Component Description	Calculated			MRP-191 Estimated	
		Fluence (n/cm^2)	Iron Atom Displacements (dpa)	Region	Region	Fluence Range (n/cm^2)
Interfacing Components	Clevis insert bolts	$< 10^{20}$	< 0.15	1	1	$< 10^{20}$
	Clevis insert dowels	$< 10^{20}$	< 0.15	1	1	$< 10^{20}$
	Clevis insert locking devices	$< 10^{20}$	< 0.15	1	1	$< 10^{20}$
	Clevis inserts	$< 10^{20}$	< 0.15	1	1	$< 10^{20}$
	Head and vessel alignment pin bolts	$< 10^{20}$	< 0.15	1	1	$< 10^{20}$
	Head and vessel alignment pin lock caps	$< 10^{20}$	< 0.15	1	1	$< 10^{20}$
	Head and vessel alignment pins	$< 10^{20}$	< 0.15	1	1	$< 10^{20}$
	Internals hold-down spring	$< 10^{20}$	< 0.15	1	1	$< 10^{20}$
	UCP alignment pins	$3x10^{20}$	0.48	2	3	$7x10^{20}$ to $1x10^{21}$
	Replacement reactor vessel head (RRVH) extension tubes	$< 10^{20}$	< 0.15	1	1	$< 10^{20}$
Note: The components located in the active fuel region were conservatively assigned to Region 6 and were not explicitly modeled.						

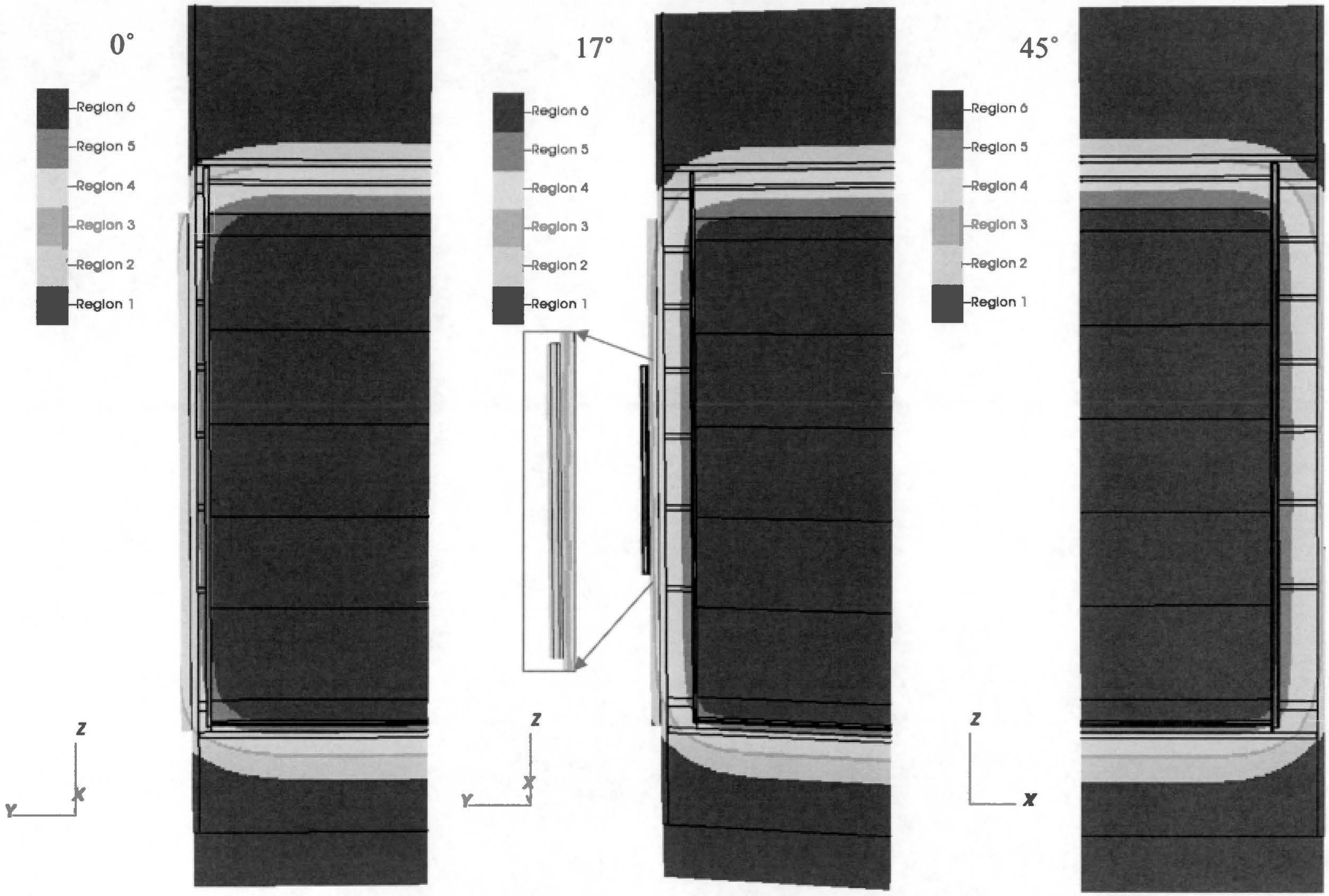

Figure 4-1. Cross-Sectional Views of the Reactor Internals at Selected Azimuthal Locations

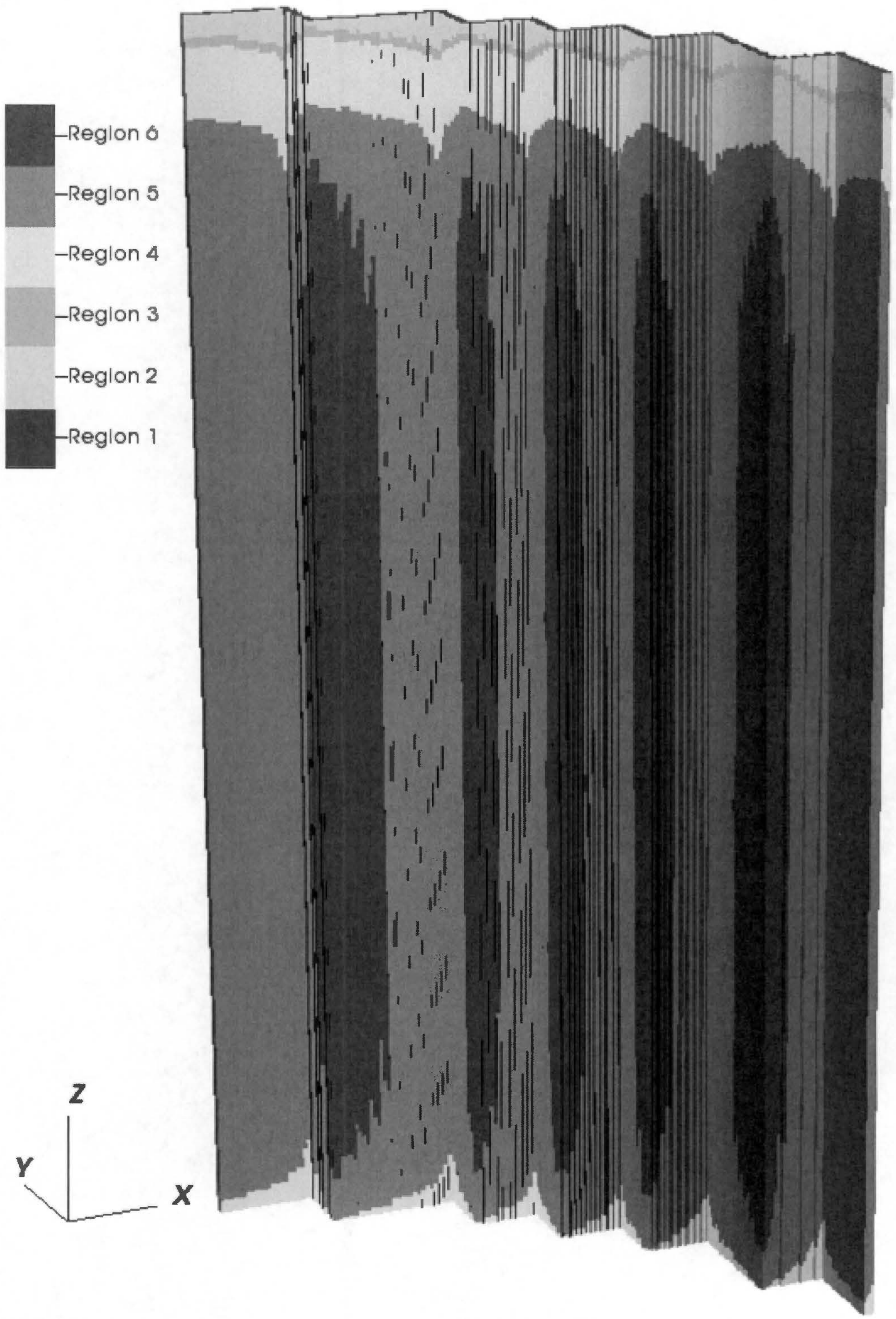

Figure 4-2. Regional Map of the Core Baffle

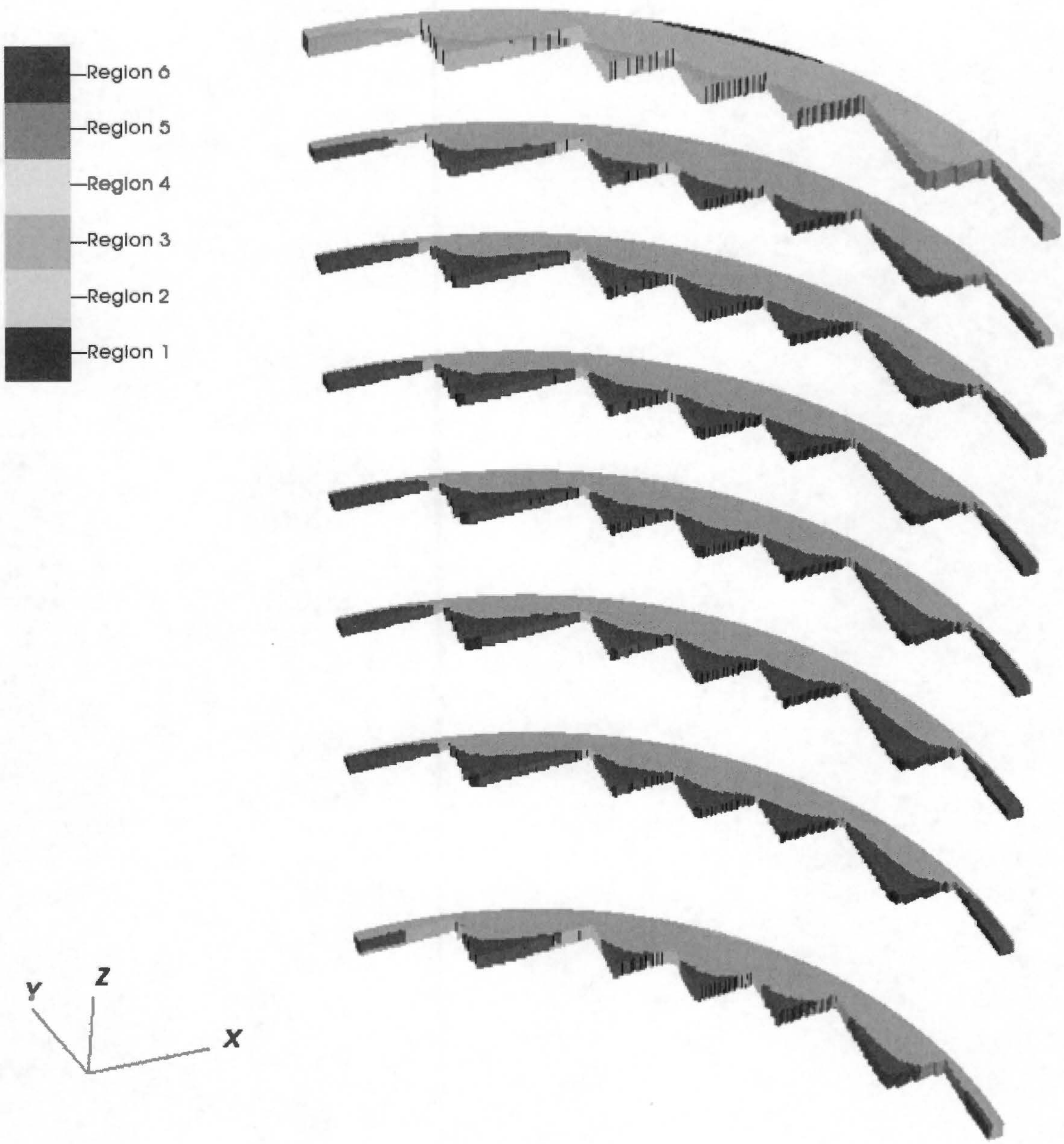

Figure 4-3. Regional Map of the Formers

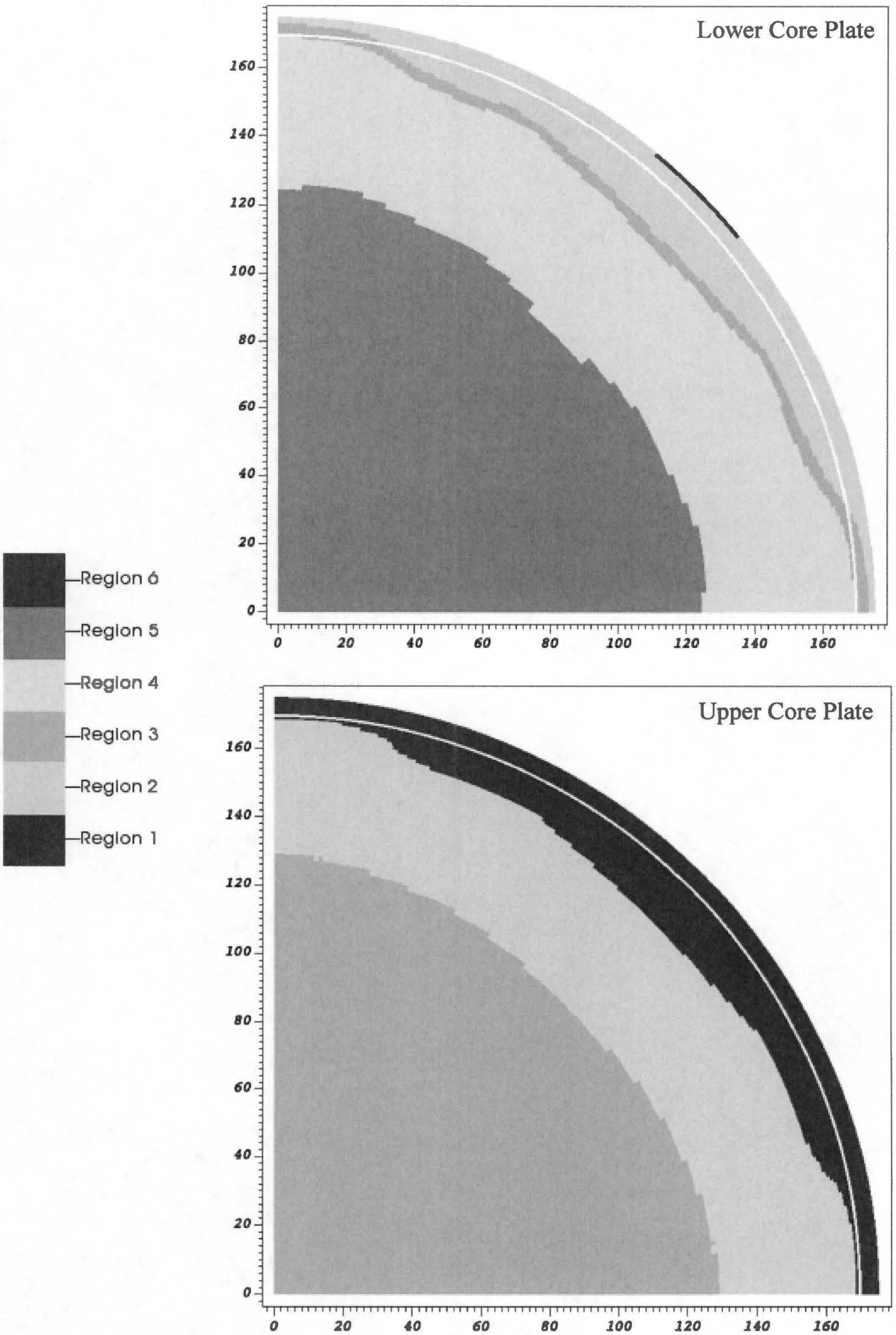

Figure 4-4. Regional Maps at Select Axial Cross-Sections of the Reactor Internals
(Dimensions are distance from core center in centimeters. Core Barrel is also shown.)

5 REFERENCES

1. Electric Power Research Institute (EPRI) Document, MRP-191, Revision 2, "Materials Reliability Program: Screening, Categorization, and Ranking of Reactor Internals Components for Westinghouse and Combustion Engineering PWR Design (MRP-191, Revision 2)," 2018.

2. USNRC Regulatory Guide 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence," Office of Nuclear Regulatory Research, March 2001.

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4. Westinghouse Report WCAP-18709-NP, Revision 1, "V.C. Summer Nuclear Station Unit 1 Subsequent License Renewal: Reactor Pressure Vessel Extended Beltline Neutron Exposure Evaluation," February 2022.