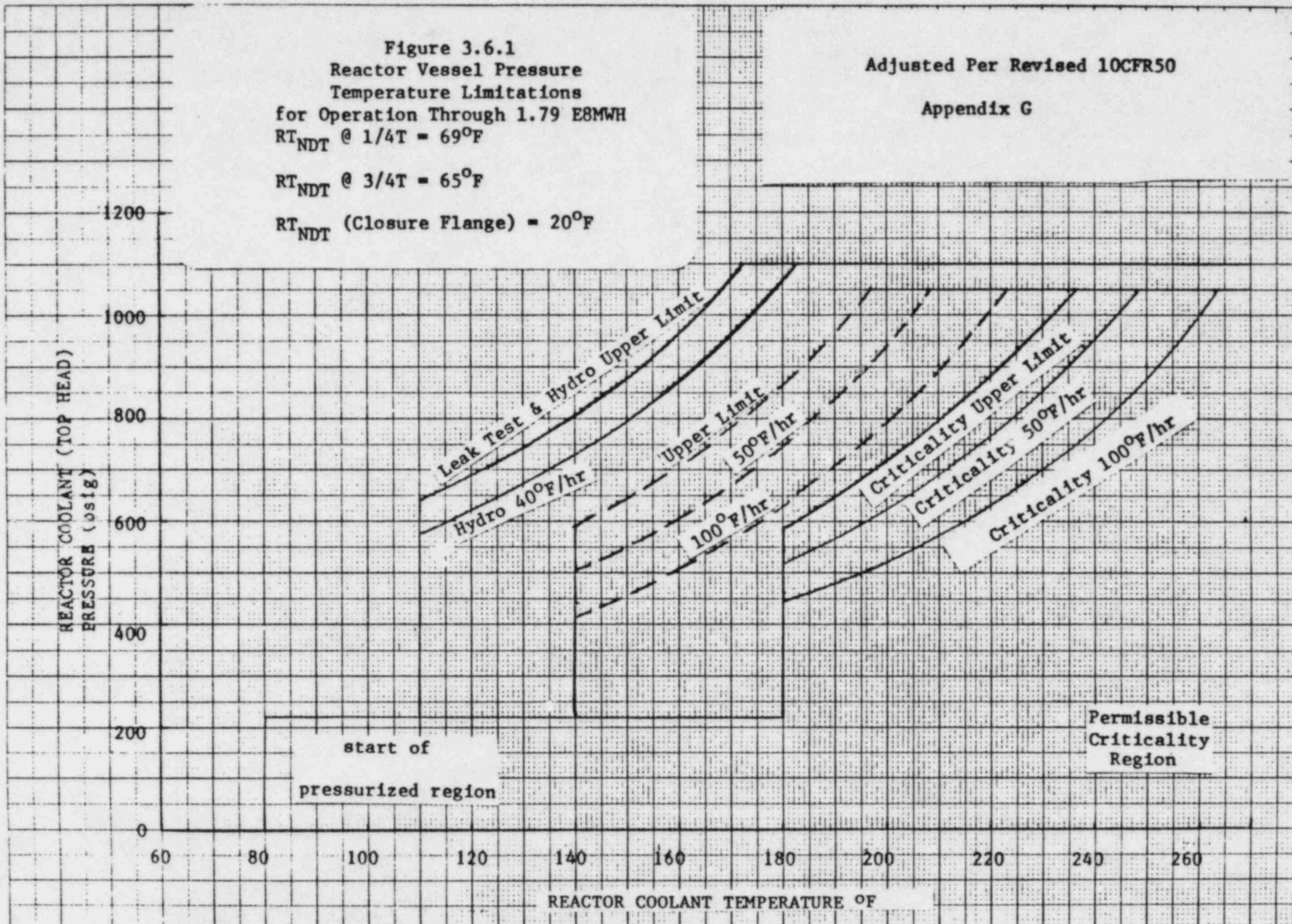
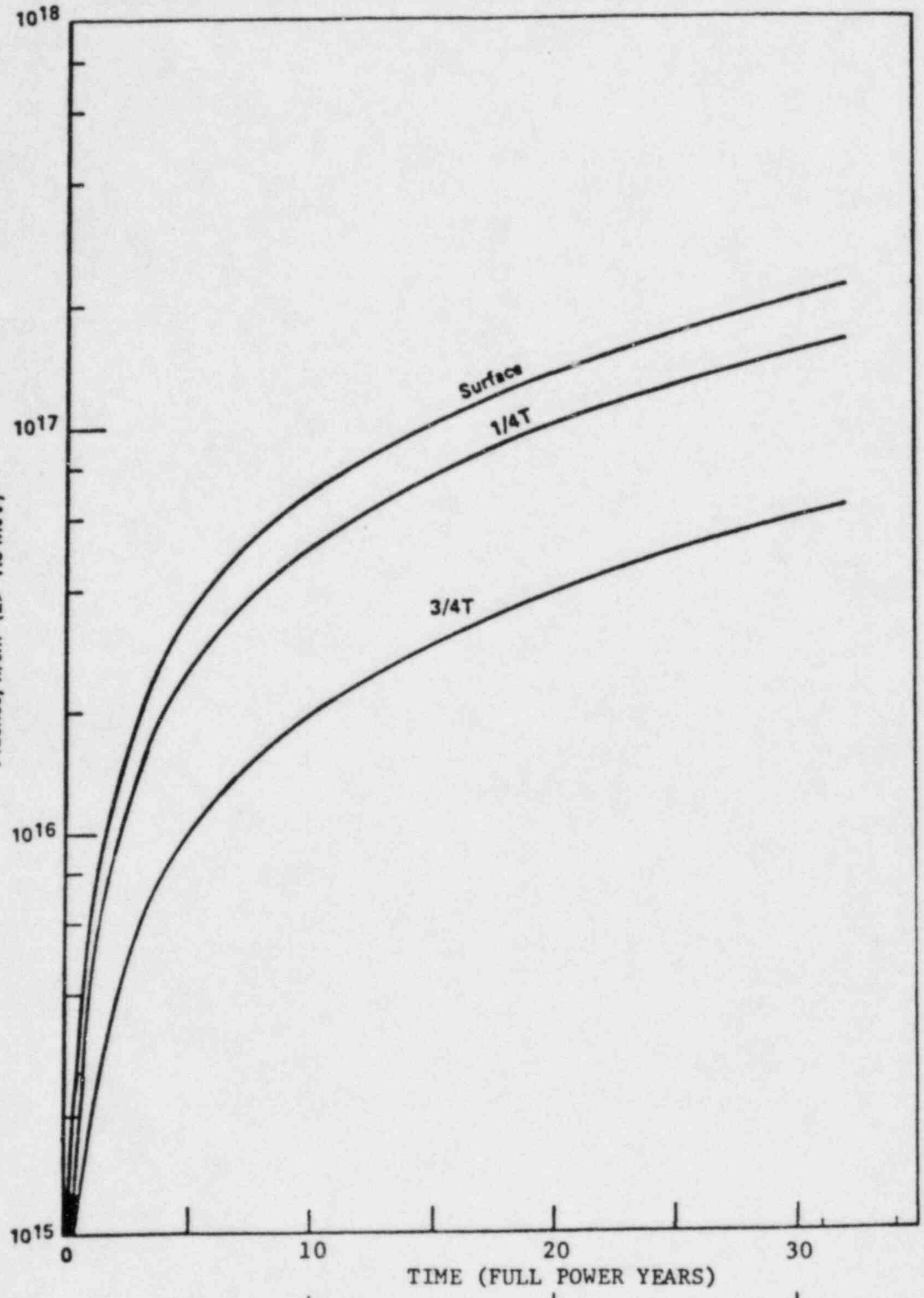


Figure 3.6.1
Reactor Vessel Pressure
Temperature Limitations
for Operation Through 1.79 E8MWH
RT_{NDT} @ 1/4T = 69°F
RT_{NDT} @ 3/4T = 65°F
RT_{NDT} (Closure Flange) = 20°F

FIGURE 3.6.2
FAST NEUTRON FLUENCE (E > 1 MEV) AS A FUNCTION OF THERMAL ENERGY AND FULL POWER YEARS



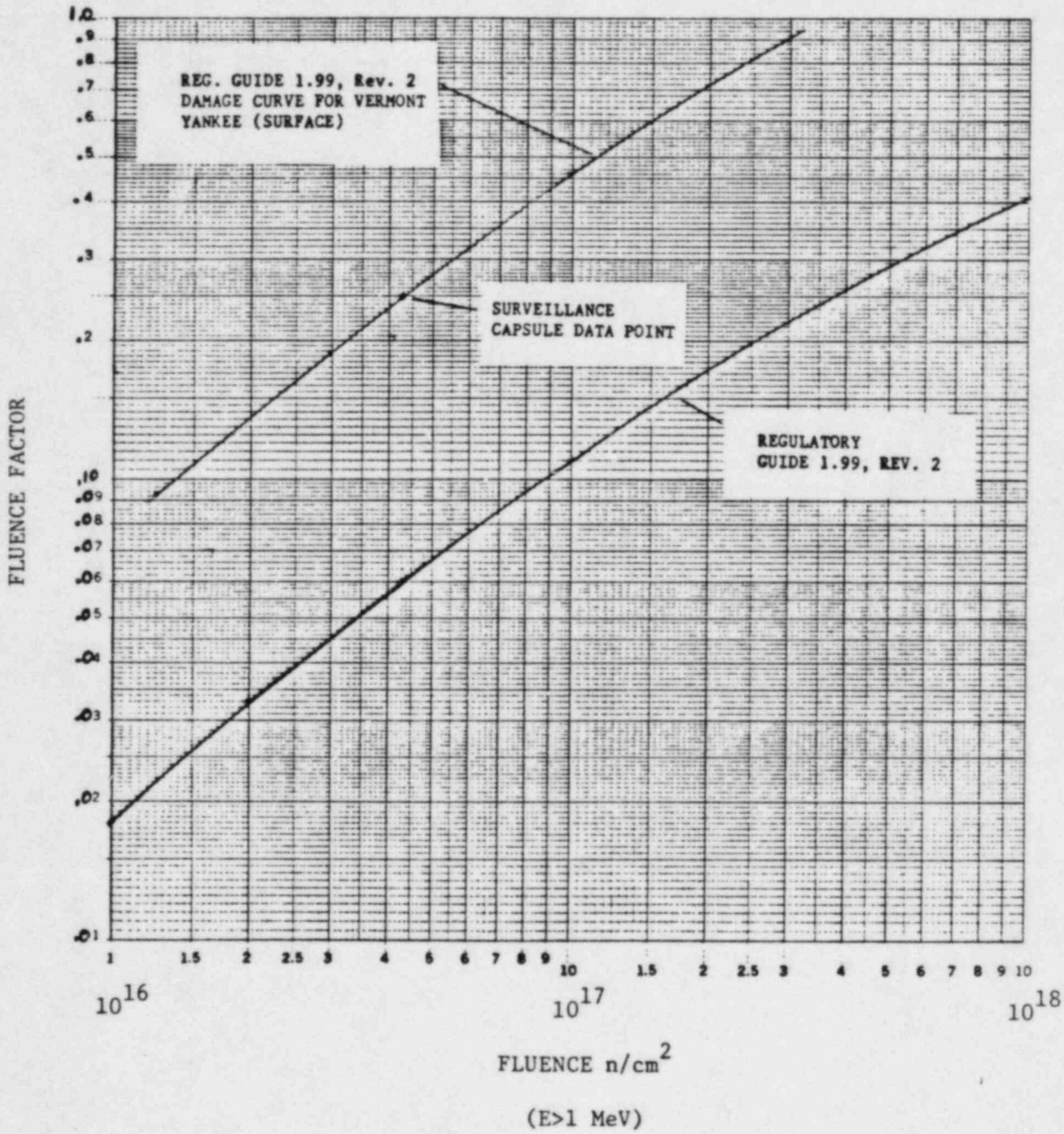
REFERENCE: L. M. Lowry et al. "Examination, Testing, and Evaluation of Irradiated Pressure Vessel Surveillance Specimens from Vermont Yankee Nuclear Power Station.

Batelle Columbus Laboratories Report #BCL-585-84-3, May 15, 1984

FIGURE 3.6.3

FLUENCE FACTOR FOR USE IN REGULATORY GUIDE 1.99

Rev. 2



Bases

3.6 and 4.6 Reactor Coolant System

A. Pressure and Temperature Limitations

All components in the Reactor Coolant System are designed to withstand the effects of cyclic loads due to system temperature and pressure changes. These cyclic loads are introduced by normal load transients, reactor trips, and startup and shutdown operations. The various categories of load cycles used for design purposes are provided in Section 4.2 of the FSAR. During startup and shutdown, the rates of temperature and pressure changes are limited so that the maximum specified heatup and cooldown rates are consistent with the design assumptions and satisfy the stress limits for cyclic operation.

During heatup, the thermal gradients in the reactor vessel wall produce thermal stresses which vary from compressive at the inner wall to tensile at the outer wall. These thermal induced compressive stresses tend to alleviate the tensile stresses induced by their internal pressure. Therefore, a pressure-temperature curve based on steady-state conditions (i.e., no thermal stresses) represents a lower bound of all similar curves for finite heatup rates when the inner wall of the vessel is treated as the governing locations.

The heatup analysis also covers the determination of pressure-temperature limitations for the case in which the outer wall of the vessel becomes the controlling location. The thermal gradients established during heatup produce tensile stresses at the outer wall of the vessel. These stresses are additive to the pressure induced tensile stresses which are already present. The thermal induced stresses at the outer wall of the vessel are tensile and are dependent on both the rate of heatup and the time along the heatup ramp; therefore, a lower bound curve similar to that described for the heatup of the inner wall cannot be defined. Subsequently, for the cases in which the outer wall of the vessel becomes the stress controlling location, each heatup rate of interest must be analyzed on an individual basis.

In order to prevent undue stress on the vessel nozzles and bottom head region, the recirculation loop temperatures should be within 50^{o}F of each other prior to startup of an idle loop.

The reactor vessel materials have been tested to determine their initial reference temperature nil-ductility transition temperature (RT_{NDT}) of 40^{o}F maximum. Reactor operation and resultant fast neutron (E >1 Mev) irradiation will cause an increase in the RT_{NDT}. Therefore, an adjusted reference temperature can be predicted using current industry practices and Vermont Yankee Surveillance Program data. (Regulatory Guide 1.99, Revision 2, and Battelle Columbus Laboratory Report BCL 585-84-3, dated May 15, 1984. The pressure/temperature limit curve, Figure 3.6.1, includes predicted adjustments for this shift in RT_{NDT} for operation through 1.79×10^{8} MWH(t), as well as adjustments for possible errors in the pressure and temperature sensing instruments.

The reference temperature of the closure flange material was determined by material testing and Branch Technical Position MTEB 5-2, "Fracture Toughness Requirements for Older Plants". The closure flange is located in a low neutron fluence area and therefore no measurable RT_{NDT} shift is expected over the life of the plant.

The actual shift in RT_{NDT} of the vessel material will be established periodically during operation by removing and evaluating, in accordance with ASTM E185-73, reactor vessel material irradiation surveillance specimens installed near the inside wall of the reactor vessel in the core area. Since the neutron spectra at the irradiation samples and vessel inside radius are essentially identical, the measured transition shift for a sample can be applied with confidence to the adjacent section of the reactor vessel. Battelle Columbus Laboratory Report BCL-585-84-3, dated May 15, 1984, provides this information for the ten-year surveillance capsule. In order to estimate the material properties at the 1/4 and 3/4 T positions in the vessel plate, the shift in RT_{NDT} is determined in accordance with Regulatory Guide 1.99, Revision 2. The heatup and cooldown curves must be recalculated when the ΔRT_{NDT} determined from the surveillance capsule is different from the calculated ΔRT_{NDT} for the equivalent capsule radiation exposure.

The pressure-temperature limit lines, shown on Figure 3.6.1, for reactor criticality and for inservice leak and hydrostatic testing have been provided to assure compliance with the minimum temperature requirements of Appendix G to 10CFR50 for reactor criticality and for inservice leak and hydrostatic testing.

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

Coolant Chemistry

A steady-state radioiodine concentration limit of 1.1 μCi of I-131 dose equivalent per gram of water in the Reactor Coolant System can be reached if the gross radioactivity in the gaseous effluents is near the limit, as set forth in Specification 3.8.C.1a, or there is a failure or prolonged shutdown of the cleanup demineralizer. In the event of a steam line rupture outside the drywell, the NRC staff calculations show the resultant radiological dose at the site boundary to be less than 30 Rem to the thyroid. This dose was

ATTACHMENT 1

Determination of Fracture Mechanics Parameters

1.0 MATERIAL PARAMETERS

In order to develop revised curves, several important material parameters need to be re-established or revised for the Vermont Yankee reactor vessel limiting material. Changes are needed to reflect the results of impact tests performed on surveillance capsule material which was removed from the Reactor Vessel in March 1983. Reference C contains results of this testing.

In addition, new tests were performed on unirradiated archival base, weld, and heat affected zone specimens to more clearly establish initial nil-ductility transition temperatures. (Reference I)

The following parameters are developed in detail.

Initial RT_{NDT}

Shift in RT_{NDT}

Vessel Surface Fluence

Adjusted RT_{NDT}

RT_{NDT} of the Closure Flange Material

The base metal for the Vermont Yankee reactor pressure vessel is A533 Grade B, Class 1 steel. Charpy V-notch and tensile specimens were prepared from an actual beltline plate (No. 2 shell and piece marked 1-14). The specimens were prepared from A533 steel plate (Heat No. C3017-2) provided by Lukens Steel Corporation in 1969.

Only two plates lie in the vessel belt line, pieces 1-14 and 1-15. Mechanical testing results indicate that piece 1-15 has clearly superior initial impact properties. In addition, the chemistry of plates 1-14 and 1-15 are nearly identical. Similar shift in RT_{NDT} for each plate would be expected.

The limiting plate is thus established as piece 1-14 which is the surveillance plate.

A. CALCULATE INITIAL RT_{NDT}

Method 1

Using ASME Code Section III (Subsection NB)

NB-2331:

(a) (4) states: "a temperature representing a minimum of 50 ft.-# and 35 mils (0.89mm) lateral expansion may be obtained from a full C_v impact curve ....."

A temperature of T_{cv} = 80°F was obtained from Battelle Report BCL-585-84-1, Table 2, Page 9, for unirradiated base metal specimens. Test results at T_{cv} are ≥ 50 ft-lb and ≥ 35 mil lateral expansion.

Then from (a) (3) the initial RT_{NDT} is calculated as T_{cv} - 60 = 80 - 60 = 20°F. This is the initial RT_{NDT} for a longitudinal Charpy specimen.

To convert longitudinal to transverse, the USNRC Materials Engineering Branch Technical Position was used. (See Standard Review Plan, NUREG-0800, dated July 1981, Revision 1, Pages 5.3.2-13 and 14, Item (3) (b).)

20 + 20 = 40°F = RT_{NDT} for transverse VY base metal

Method 2

Using Standard Review Plan, Page 5.3.2-14, Item (1):

If drop weight 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 30 ft-lbs was obtained in Charpy V-notch tests, or 0°F whichever is higher.

From the Battelle Report BCL-585-84-1, Table 2, Page 9, for unirradiated base metal specimens for Vermont Yankee:

46.5 ft-lb is obtained at 40°F

Using the Standard Review Plan, Item (3) (a) to obtain transverse properties, take 65% of 46.5 ft-lb:

(46.5 ft-lb) (.65) = 30.2 ft-lb at 40°F

Therefore, 40°F is a conservative estimate of the RT_{NDT} for transverse VY base metal.

Method 3

Utilizing Chicago Bridge and Iron Company, Drawing 5920, Revision 2, and Standard Review Plan 5.3.2-14, Charpy tests were originally run at one temperature and drop weight data was determined from longitudinally oriented specimens.

For Pc. Mark 1-14, Heat #C3017

Charpy Data

@ + 40°F	72.5 ft-#)	Obtained From
	65 ft-#)	Three Specimens
	82 ft-#)	

Drop Weight

No break @ + 20°F

Applying SRP 5.3.2, Page 14, Paragraph (4):

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

From the data above:

RT_{NDT} = 40°F

Thus, an initial RT_{NDT} = 40°F is established by three methods.

B. DETERMINATION OF SHIFT IN RT_{NDT}

From the Battelle Report (Reference C), the shift in RT_{NDT} was 19°F at a fluence of 4.3×10^{16} n/cm^2. Utilizing draft Regulatory Guide 1.99, Revision 2, a shift of only 4.7°F results at this fluence. The Vermont Yankee Chemistry Factor (CF) is 76, representing a copper content of 0.11 weight percent and a nickel content of 0.68 weight percent. The measured shift is within one standard deviation of the calculated (Regulatory Guide 1.99, Revision 2 assumes 1 = 18° for base metal, see Paragraph C). However, since the Regulatory Guide results in an unconservative prediction of shift, a modified Regulatory Guide curve was developed. The modified curve utilizes the same curve shape and damage prediction as Regulatory Guide 1.99, Revision 2, but passes through the Vermont Yankee surveillance capsule data point. In effect, the fluence factor parameter in the Regulatory Guide 1.99 shift equation is multiplied by a factor of 4.17 to duplicate the measured RT_{NDT} shift at Vermont (see Paragraph D). Future shift values can then be determined from this curve until the next surveillance specimen is removed.

C. REGULATORY GUIDE 1.99, REVISION 2 PREDICTION OF ΔRT_{NDT} FOR THE SURVEILLANCE CAPSULE MATERIAL

$$\Delta RT_{NDT_{SUR}} = [CF]f^{(0.28 - 0.1 \log f)}$$

CF = 76 @ Cu = 0.11   Reference C
Ni = 0.68   Page 3

At time of capsule removal:

flu = 4.3×10^{16} or $.0043 \times 10^{19}$

fluence factor = $(.0043)^{(.28 - 0.1 \log .0043)}$

= .060

ΔRT_{NDT} (76)(.060) = 4.6°F

Margin = lesser of $2\sqrt{\sigma_I^2 + \sigma_\Delta^2}$ or $(0.5)(\Delta RT_{NDT})$

= 2.3°

ΔRT_{NDT} = 4.6 + 2.3 = 6.9°F

This value is less than the measured shift of 19°, but within one standard deviation (1σ) which is defined as 18° by the Regulatory Guide;

i.e., measured = 19° < 6.9 + 18 = 25°F

D. DETERMINATION OF REGULATORY GUIDE 1.99, REVISION 2, MULTIPLIER FOR APPLICATION TO VERMONT YANKEE SURVEILLANCE DATA

- From Battelle Report (Reference C)

  @ fluence of 4.3×10^{16}, ΔRT_{NDT} = 19°F.

- Regulatory Guide shift equation

  $\Delta RT_{NDT} = [CF]f^{(0.28 - 0.1 \log f)}$

  where CF is chemistry factor = 76 for Vermont Yankee

- Fluence factor = f (0.28 - 0.1 log f)

  where f is surface fluence/10^{19}

- Fluence factor is plotted in Regulatory Guide Figure 1

Let $\Delta RT_{NDT} = 19^{o}$ and $f = \dfrac{4.3 \times 10^{16}}{10^{19}} = .0043$

$$\therefore \frac{\Delta RT_{NDT}}{[CF]} = \frac{19}{76} = 0.25$$

$\therefore$ fluence factor for Vermont Yankee is 0.25 at a surface fluence of

$.0043 \times 10^{19}$

Fluence factor by Regulatory Guide 1.99 = .060 (see Paragraph C).

$\therefore$ multiplier is 0.25/0.060 = 4.167.

Generate modified fluence factor curve using this multiplier on Regulatory Guide curve.

Fluence	Regulatory Guide 1.99 Fluence Factor	Modified Vermont Yankee Fluence Factor
2×10^{16}	.033	.138
10^{17}	.11	.458
2×10^{17}	.175	.729

Resulting curve is shown as Proposed Figure 3.6.3 in the Technical Specifications.

E. CONVERT MWH_{th} TO SURFACE FLUENCE

At the time of the surveillance capsule removal, March 4, 1983, shutdown:

7.54 EFPY =

1.05×10^{8} $MWHr_{th}$ =

5.18×10^{16} n/cm^{2} = Surface fluence at 0^{o} azimuth (maximum) (Reference C)

Appendix G curves are calculated for following dates:

March 1986

$MWHr_{th} = 1.33 \times 10^{8}$

Surface fluence = $\frac{1.33}{1.05}(5.18 \times 10^{16}) = 6.56 \times 10^{16}$ n/cm^2

(Curves are calculated here to compare to current operating limits.)

December 1989

$MWHr_{th} = 1.79 \times 10^8$

Surface fluence = $\frac{1.79}{1.05}(5.18 \times 10^{16}) = 8.83 \times 10^{16}$ n/cm^2

End of Life

32 EFPY = 40 calendar years at 80% capacity

Surface fluence = 2.3×10^{17} (Reference C)

See Proposed Technical Specification Figure 3.6.2.

TABLE 1

SUMMARY OF CALCULATIONS FOR ADJUSTED RT_{NDT} OF VESSEL SURVEILLANCE PLATE

	12-89
FLUENCE(1) (Surface)	8.6×10^{16}
RT_{NDT} (2) (Surface)	31.3^{o}
RT_{NDT} (3) (1/4 T)	28.9^{o}
RT_{NDT} (3) (3/4 T)	24.4^{o}
RT_{NDT} (i)(4)	40^{o}
ART (1/4)(2)	68.9^{o}
ART (3/4)	64.4^{o}

---

(1) Reference C

(2) Regulatory Guide 1.99, Revision 2, modified by Vermont Yankee shift data:

$$RT_{NDT} = [76]\ (4.17)\ [f^{(.28\ -\ .1\ \log f)}] \qquad f = \frac{\text{fluence}}{10^{19}}$$

(3) Regulatory Guide 1.99, Revision 2:

RT_{NDT} (@ 1/4 or 3/4 T) = RT_{NDT} surface.$e^{-0.065x}$

$x_{(1/4)}$ = 5.06/4 = 1.265 inches, factor = 0.921

$x_{(3/4)}$ = 5.06 (3/4) = 3.795 inches, factor = 0.781

(4) Paragraph A

(5) ART(X) = $RT_{NDT}(i)$ + $RT_{NDT}(X)$

F. DETERMINE RT_{NDT} OF CLOSURE FLANGE MATERIAL

Data Sources:

(A) LADISH CO Test Report for Parts 227-3/4 inches x 199 x 28-7/16 inches to Specification MS-2, Revision 1.

(B) LADISH CO Test Report for Part 227-3/4 inches x 206-1/2 x 26-1/16 to Specification MS-2, Revision 1.

(C) SRP 5.3.2, Page 13.

From Standard Review Plan 5.3.2.13:

> "The NDTT temperature as determined by drop weight tests is the RT_{NDT} if, at 60^{o} above the NDTT at least 50 ft-# of energy and 35 mils lateral expansion are obtained in Charpy V tests ..."

From drop weight tests there was "no break at $20^{o}F$" utilizing 4 specimens from each forging.

Let NDT = $20^{o}F$ (conservative by ASTM E-208).

At $60^{o} + 20^{o} = 80^{o}$, 50 ft-# of energy and 35 mils lateral expansion are required.

From the Ladish data these values were satisfied at $+10^{o}$.

$\therefore$ RT_{NDT} (closure flange) = NDT = $+20^{o}F$.

TABLE 2

SUMMARY OF DATA

Parameter/Date	3-83	5-86[a]	12-89	EOL[b]	Prior Source (c) of Data for Existing RT_{NDT} Limits	Revised Source of Data (c)
1. $MWHr_{th} \times 10^8$	1.05	1.33	1.788	4.46	A	A
2. Effective Full Power Years	7.54	9	12.8	32	A	A
3. Fluence (Surface) n/cm^2	5.19×10^{16}	6.57×10^{16}	8.6×10^{16}	2.3×10^{17}	B	C, J
4. Fluence (1/4 T) n/cm^2	3.78×10^{16}				B	C, J
5. Fluence (3/4 T) n/cm^2	1.48×10^{16}				B	C, J
6. Initial RT_{NDT} Plate (PC MK 1-14, Heat #3017-2)	40^o	40^o	40^o	40^o	D, F	D, I, F
7. Shift in RT_{NDT} (Plate) Adjusted RT_{NDT} (6. + 7.)		23.8 63.8	28.8 68.8	54.5 94.5	K	C, E
8. Closure Flange RT_{NDT}	20^o	20^o	20^o	20^o	F	G, H, F
9. Calculational Method	N/A	N/A	N/A	N/A	L, M	L, M

a. Current pressure temperature curve limit date.

b. 80% full power operation for 40 calendar years.

c. Data sources listed in Paragraph G.

G. DATA SOURCES (REFERENCES)

A. Vermont Yankee Reactor Engineering Department.

B. General Electric Company SIL #14.

C. Battelle Columbus Laboratory Report #BCL-585-84-3, dated May 15, 1984.

D. Chicago Bridge and Iron Company, Drawing #9-6201 R-7, Revision 2.

E. USNRC Regulatory Guide 1.99, Revision 2 (not issued).

F. USNRC Mechanical Engineering Branch Technical Position MTEB 5.2.

G. Chicago Bridge and Iron Company, Drawing #9-6201 R-12, Revision 2.

H. Ladish Company Material Analysis Reports for CB&I Parts # 9-6201, M1-5 MRKD, 1-9 and 9-6201, M2-4 MRKD, 1-8.

I. Battelle Columbus Laboratory Reports #BCL-585-84-1, dated March 21, 1984.

J. Southwest Research Institute Report 02-4032, by E. B. Norris, dated May 23, 1975.

K. Vermont Yankee Nuclear Power Station FSAR.

L. ASME Boiler and Pressure Vessel Code, Section III, Appendix G.

M. 10CFR50 Appendix G. 1984 Edition.

2.0 CALCULATION OF APPENDIX G CURVES

Calculations used to develop the revised Appendix G curves, Figure 3.6.1, were performed in accordance with the requirements of 10CFR50 Appendix G (1984), ASME Boiler and Pressure Vessel Code, Section III, Appendix G (1980 Edition through Summer 1982 Addenda) and Standard Review Plan 5.3.2.