Attachment III to JPN-99-021

MARKED-UP TECHNICAL SPECIFICATION PAGES

PRESSURE-TEMPERATURE LIMITS

(JPTS-99-003)

NOTE: Deletions are shown in ~~strikeout~~, and additions are in **bold**.

New York Power Authority

JAMES A. FITZPATRICK NUCLEAR POWER PLANT
Docket No. 50-333
DPR-59

LIST OF FIGURES

Amendment No. ~~14, 22, 43, 64, 72, 74, 88, 98, 109, 113, 116, 117, 134, 137, 158, 162, 227, 236, 247~~,

3.6 LIMITING CONDITIONS FOR OPERATION

3.6 REACTOR COOLANT SYSTEM

Applicability:

Applies to the operating status of the Reactor Coolant System.

Objective:

To assure the integrity and safe operation of the Reactor Coolant System.

Specification:

A. Pressurization and Thermal Limits

1. Reactor Vessel Head Stud Tensioning

The reactor vessel head bolting studs shall not be under tension unless the temperatures of the reactor vessel flange and the reactor head flange are at least 90°F.

2. In-Service Hydrostatic and Leak Tests

During in-service hydrostatic or leak testing the Reactor Coolant System pressure and temperature shall be on or to the right of curve A shown in Figure 3.6-1 Part 1~~,~~ **or** 2 ~~, or 3 and~~ **for the flange and the beltline region, and on or to the right of curve A_{NB} for the non-beltline regions, and on or to the right of curve A_{BH} for the bottom head region.** The maximum temperature change during any one hour period shall be:

4.6 SURVEILLANCE REQUIREMENTS

4.6 REACTOR COOLANT SYSTEM

Applicability:

Applies to the periodic examination and testing requirements for the Reactor Coolant System.

Objective:

To determine the condition of the Reactor Coolant System and the operation of the safety devices related to it.

Specification:

A. Pressurization and Thermal Limits

1. Reactor Vessel Head Stud Tensioning

When in the cold condition, the reactor vessel head flange and the reactor vessel flange temperatures shall be recorded:

a. Every 12 hours when the reactor vessel head flange is ≤120°F and the studs are tensioned.

b. Every 30 minutes when the reactor vessel head flange is ≤100°F and the studs are tensioned.

c. Within 30 minutes prior to and every 30 minutes during tensioning of reactor vessel head bolting studs.

2. In-Service Hydrostatic and Leak Tests

During hydrostatic and leak testing the Reactor Coolant System pressure and temperature shall be recorded every 30 minutes until two consecutive temperature readings are within 5°F of each other.

3.6 (cont'd)

a. ≤20°F when to the left of curve C.

b. ≤100°F when on or to the right of curve C.

Specifications 3.5.C, 3.5.D, 3.5.E and 3.6.E which would become effective because of an increase in reactor coolant temperature above 212°F or pressures above 100 and 150 psig are not required while conducting the RCS hydrostatic pressure and leakage tests between 212°F and 300°F provided all control rods are fully inserted.

3. Non-Nuclear Heatup and Cooldown

During heatup by non-nuclear means (mechanical), cooldown following nuclear shutdown and low power physics tests the Reactor Coolant System pressure and temperature shall be on or to the right of the curve B shown in Figure 3.6-1 Part 1~~,~~ **or** 2~~, or 3 and~~ **for the flange, upper vessel and beltline regions, and on or to the right of curve B_{BH} for the bottom head region.** The maximum temperature change during any one hour shall be ≤100°F.

4. Core Critical Operation

During all modes of operation with a critical core (except for low power physics tests) the Reactor Coolant System pressure and temperature shall be at or to the right of the curve C shown in Figure 3.6-1 Part 1~~,~~ **or** 2~~, or 3 and~~. The maximum temperature change during any one hour shall be ≤100°F.

4.6 (cont'd)

3. Non-Nuclear Heatup and Cooldown

During heatup by Non-Nuclear means, cooldown following nuclear shutdown and low power physics tests, the reactor coolant system pressure and temperature shall be recorded every 30 minutes until two consecutive temperature readings are within 5°F of each other.

4. Core Critical Operation

During all modes of operation with a critical core (except for low power physics tests) the Reactor Coolant System pressure and temperature shall be recorded within 30 minutes prior to withdrawal of control rods to bring the reactor critical and every 30 minutes during heatup until two consecutive temperature readings are within 5°F of each other.

3.6 and 4.6 BASES

A. Pressurization and Thermal Limits

The reactor vessel design specification requires that the reactor vessel be designed for a maximum heatup and cooldown rate of the contained fluid (water) of 100°F/hr averaged over a period of 1 hour. This rate has been chosen based on past experience with operating power plants. The associated time periods for heatup and cooldown cycles when the 100°F/hr rate is applied provide for efficient, but safe, plant operation.

The reactor vessel manufacturer has designed the vessel to the above temperature criterion. In the course of completing the design, the manufacturer performed detailed stress analysis. This analysis includes more severe thermal conditions than those which would be encountered during normal heating and cooling operations.

Specific analyses were made based on a heating and cooling rate of 100°F/hr applied continuously over a temperature range of 100°F to 546°F.

Calculated stresses were within 1965 ASME Boiler and Pressure Vessel Code, Section III, with 1966 addenda stress intensity and fatigue limits. The normal heating and cooling rate of 100°F/hr was also evaluated to assure protection against brittle fracture of the vessel shell remote from discontinuities. The rate meets the requirements of Appendix G to the Summer 1972 Edition of 1971 ASME III, throughout plant life, and is, therefore, satisfactory.

The limiting coolant temperature differential between the upper and lower regions of the reactor vessel, prior to recirculation pump operation, assures that the vessel bottom head region will not be warmed at an excessive rate due to rapid sweep-out of cold coolant in the vessel lower head region by recirculation pump operation (cold coolant can accumulate as a result of control rod drive inleakage and/or low recirculation flow rate during startup or hot standby). The limit on idle recirculation loop startup avoids high thermal stress effects in the pumps and piping, while also minimizing thermal stresses on the vessel nozzles.

The nil-ductility transition (NDT) temperature RT_{NDT} is defined as the temperature below which ferritic steel breaks in a brittle rather than ductile manner. Reactor vessel flux monitoring samples are installed to conform with the 1972 draft revision of ASTM E 185. Surveillance program test results have established the magnitude of changes in the NDT temperature as a function of the integrated neutron exposure for BWR vessels. The design life of the reactor vessel is 40 years, and the maximum fast neutron exposure at 40 years was originally calculated to be 7.0×10^{17} n/cm². Based on the surveillance program test results, the EOL fluence is now estimated to be ~~1.7~~ **1.8** x 10^{18} n/cm².

Fast neutron irradiation affects the fracture toughness of the reactor vessel material. In order to assure that non-ductile failure does not occur, two types of information are needed:

a) a relationship between the change in RT_{NDT} and the accumulated fast neutron fluence, and,

b) a relationship between the neutron fluence at the point of peak flux in the reactor pressure vessel shell and the effective full power years.

3.6 and 4.6 BASES (cont'd)

The expected neutron fluence at the reactor vessel wall can be determined at any point during plant life based on the linear relationship between the reactor thermal power output and the corresponding number of neutrons produced. Accordingly, neutron flux wires were removed from the reactor vessel with the surveillance specimens to establish the correlation at the capsule location by experimental methods. The flux distribution at the vessel wall and 1/4 thickness (1/4T) depth was analytically determined as a function of core height and azimuth to establish the peak flux location in the vessel and the lead factor of the surveillance specimens.

Regulatory Guide 1.99, Revision 2 **(May 1988)** is used to predict the shift in RT_{NDT} as a function of fluence in the reactor vessel beltline region. An evaluation **of two sets** of the irradiated surveillance specimens, which were withdrawn from the reactor in April, 1985 (6 EFPY) **and November 1996 (13.4 EFPY), respectively**, shows a shift in RT_{NDT} less than that predicted by Regulatory Guide 1.99, Revision 2.

Operating limits for the reactor vessel pressure and temperature during normal heatup and cooldown, and during in-service hydrostatic and leak testing were established using 10 CFR 50 Appendix G, **December 1995**, ~~May, 1983~~ and Appendix G of **Section XI** ~~the Summer 1984 Addenda to Section III~~ of the ASME Boiler and Pressure Vessel Code **(1989 Edition)**. These operating limits assure that the vessel could safely accommodate a postulated surface flaw having a depth of 0.24 inch at the flange-to-vessel junction, and one-quarter of the material thickness at all other reactor vessel locations and discontinuity regions. For the purpose of setting these operating limits, the reference temperature, RT_{NDT}, of the vessel material was estimated from impact test data taken in accordance with the requirements of **Section III of the ASME Boiler and Pressure Vessel Code** ~~the Code~~ to which the vessel was designed and manufactured (1965 Edition including Winter 1966 addenda). ~~The RT_{NDT} values for the reactor vessel flange region and for the reactor vessel shell beltline region are 30°F, based on fabrication test reports.~~

~~The RT_{NDT} for the remainder of the vessel is 40°F.~~ **(Insert A).**

The actual shift in the RT_{NDT} of the vessel material will be established periodically by removing and evaluating flux monitoring surveillance capsules in accordance with ASTM E 185-82 and 10 CFR 50, Appendix H. The evaluation findings and recommendations of Regulatory Guide 1.99 Revision 2 will provide the basis for revising Figure 3.6-1 curves A, B and C for operation of the plant. The first surveillance capsule containing test specimens was withdrawn in April, 1985 after 6 EFPY. The test specimens removed were tested according to ASTM E 185-82 and the results are in GE report MDE-49-0386. **(Insert B)** The NRC approved schedule for subsequent specimen withdrawal is located in the updated FSAR (Section 4.2.7).

~~Figure 3.6-1 is comprised of three parts: Part 1, Part 2, and Part 3. Parts 1, 2, and 3 establish the pressure temperature limits for plant operations through 12, 14, and 16 Effective Full Power Years (EFPY) respectively. The appropriate figure and the pressure temperature curves are dependent on the number of accumulated EFPY. Figure 3.6-1, Part 1 is for operation through 12 EFPY, Figure 3.6-1, Part 2 is for operation at greater than 12 EFPY through 14 EFPY, and Figure 3.6-1, Part 3 is for operation at greater than 14 EFPY through 16 EFPY. The curves contained in Figure 3.6-1 are developed from the General Electric Report DRF 137-0010, "Implementation of Regulatory Guide 1.99, Revision 2 for the James A. FitzPatrick Nuclear Power Plant," dated June, 1989.~~ **(Insert C)**

Figure 3.6-1 curve**s** A, $\mathbf{A_{BH}}$**, and** $\mathbf{A_{NB}}$ establish**es** the minimum temperature for hydrostatic and leak testing required by the ASME Boiler and Pressure Vessel Code, Section XI. Test pressures for in-service hydrostatic and leak testing are a function of the testing temperature and the component material. Accordingly, the maximum hydrostatic test pressure will be 1.1 times the operating pressure or about 1,144 psig.

Insert A

The RT_{NDT} values of the reactor vessel materials are listed on Table 3-2 of General Electric Report GE-NE-B1100732-01, "Plant FitzPatrick RPV Surveillance Materials Testing and Analysis of 120° Capsule at 13.4 EFPY," Revision 1 (February 1998), including Errata and Addenda dated June 1999.

Insert B

The second surveillance capsule containing test specimens was withdrawn in November, 1996 after 13.4 EFPY. The test specimens removed were tested according to ASTM E 185-82 and the results are in General Electric Report GE-NE-B1100732-01, Revision 1 (February 1998), including Errata and Addenda dated June 1999.

Insert C

Figure 3.6-1 is comprised of two parts: Part 1 and Part 2. Part 1 establishes the pressure-temperature limits for the bottom head, flange, upper vessel and beltline regions for plant operations through 24 Effective Full Power Years (EFPY). Part 2 establishes the pressure-temperature limits for plant operations through 32 EFPY. The curves contained in Figure 3.6-1 are developed from the General Electric Report GE-NE-B1100732-01, Revision 1 (February 1998), including Errata and Addenda dated June 1999.

3.6 and 4.6 BASES (cont'd)

Fig. 3.6-1, curves B **and B_{BH}**, provides limitations for plant heatup and cooldown when the reactor is <u>not critical</u> or during low power physics tests. The thermal limitation is based on maximum heatup and cooldown rates of 100°F/hr in any one-hour period.

Fig. 3.6-1, curve C, establishes operating limits when core is critical. These limits include a margin of 40°F as required by 10 CFR 50 Appendix G.

The requirements for cold boltup of the reactor vessel closure **region** are **established** based on ~~NDT~~ RT_{NDT} ~~temperature~~ plus ~~a~~ 60°F ~~factor of safety~~. This factor is based on the **original** requirements of the ASME Code to which the vessel was built, **as well as additional conservative requirements developed by General Electric that are typically applied to most BWRs**. For Fig. 3.6-1, curves A, B, and C, **this factor leads to the 90°F lower temperature limit**.~~margins are only added to the low temperature portion of the curve where non ductile failure is a concern.~~ **This is based on** the closure flange~~s have an NDT temperature not greater than~~ **materials maximum RT_{NDT} of** 30°F and **the fact that the closure flange materials** are not subject**ed** to any appreciable neutron radiation exposure. Therefore, the minimum temperature **for cold boltup** ~~of the flanges when the studs are in tension~~ is 30°F plus 60°F, or 90°F.

Specification 3.6.A.2 identifies four LCOs that become effective with increased reactor coolant temperature or pressure but are not in effect during the hydrostatic and leakage tests. This is necessary because, as reactor fluence increases, the minimum test temperature and pressure rises into ranges normally associated with startup or hot shutdown. RCS pressure and temperature are used throughout the Technical Specifications as a basis for establishing plant mode and system operability requirements.

Some LCOs and restrictions cannot be satisfied during the test at elevated temperatures. For example, Specifications 3.5.C.1 and 3.5.E.1 require that HPCI and RCIC be operable when reactor pressure exceeds 150 psig and 212°F. HPCI and RCIC cannot be made operable during the test because piping normally filled with steam is filled with water during the test.

Hydrostatic and leakage tests shall be terminated before the reactor coolant temperature exceeds 300°F. This temperature limit is based on providing **at least** a 50°F band for operating flexibility between the 300°F limit and the highest estimated minimum testing temperature at 32 EFPY (**originally** approximate~~ly~~**d as** 250°F **from testing of the first surveillance capsule**). **Based on the latest surveillance capsule test results, the minimum temperature required to stay on or to the right of curve A at the maximum test pressure is 212°F for 32 EFPY. The previously established hydrostatic test limitation of 300°F continues to provide adequate operating flexibility above this minimum temperature.**

The protection provided by LCOs applicable during cold shutdown plus the requirement that all control rods be fully inserted are adequate to ensure protection of public health and safety. The hydrostatic test is performed once every 10 years while the leakage test is performed after each refueling when conditions are similar to cold shutdown (i.e., after the reactor has been shutdown and decay heat and the energy stored in the core is very low). The consequences of accidents (small and large break LOCAs, MSLB, etc.) are bounded by analyses that assume full power operation. Specification 3.5.A requires the low pressure ECCS systems to be operable. Specifications 3.7.A, 3.7.B and 3.7.C require the containment, SGTS and secondary containment to be operable. Specifications 3.2.A, 3.2.B and Appendix B, Specification 3.8 require instrumentation that initiate containment, low pressure ECCS, SBGT and secondary containment be operable. Emergency power is required by Specification 3.9.B.

INSERT D

~~Figure 3.6-1 Part 1 Reactor Vessel Pressure - Temperature Limits Through 12 EFPY~~

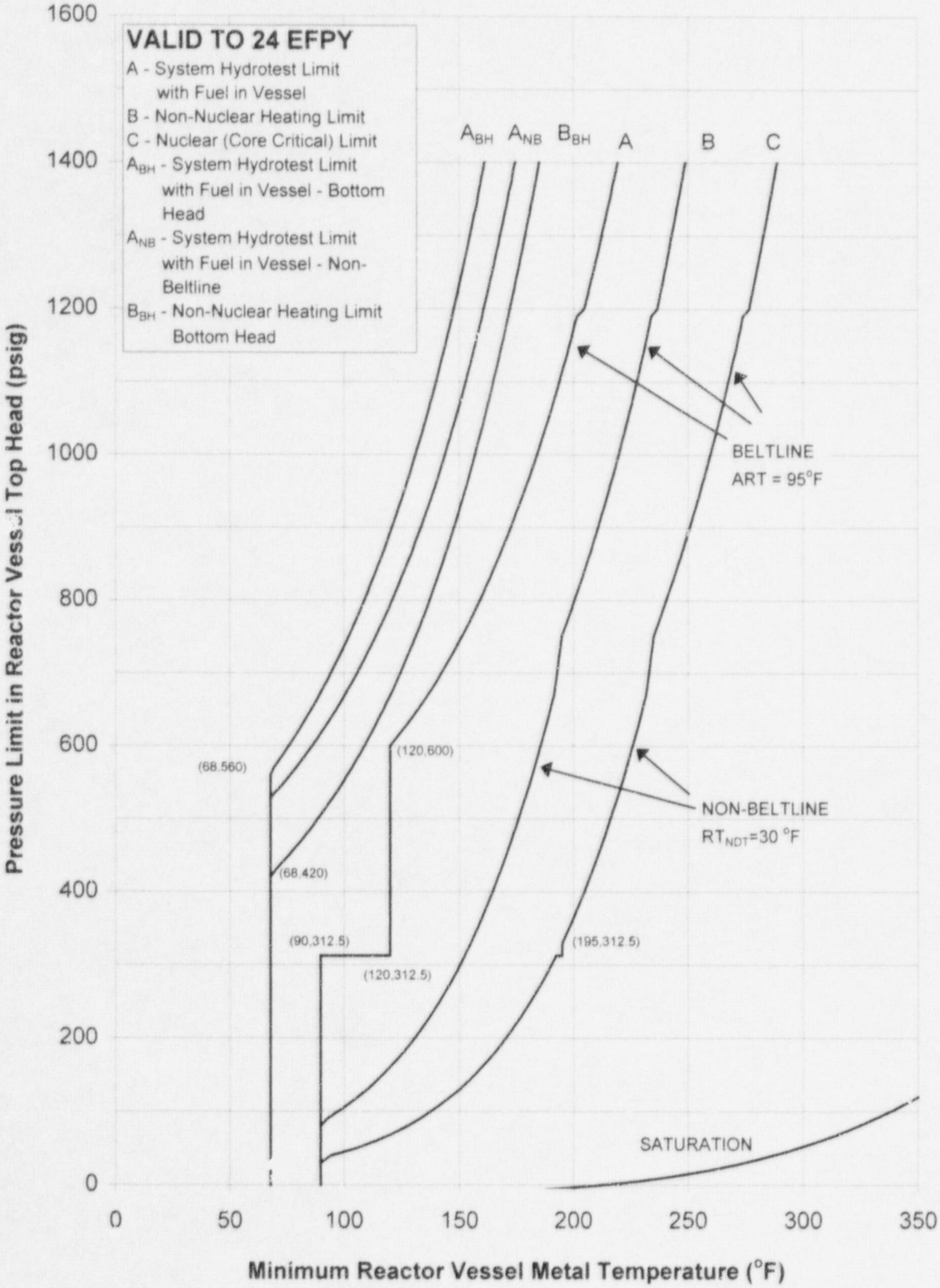

Figure 3.6-1 Part 1 Reactor Vessel Pressure-Temperature Limits Through 24 EFPY

INSERT E

~~Figure 3.6-1 Part 2 Reactor Vessel Pressure - Temperature Limits Through 14 EFPY~~

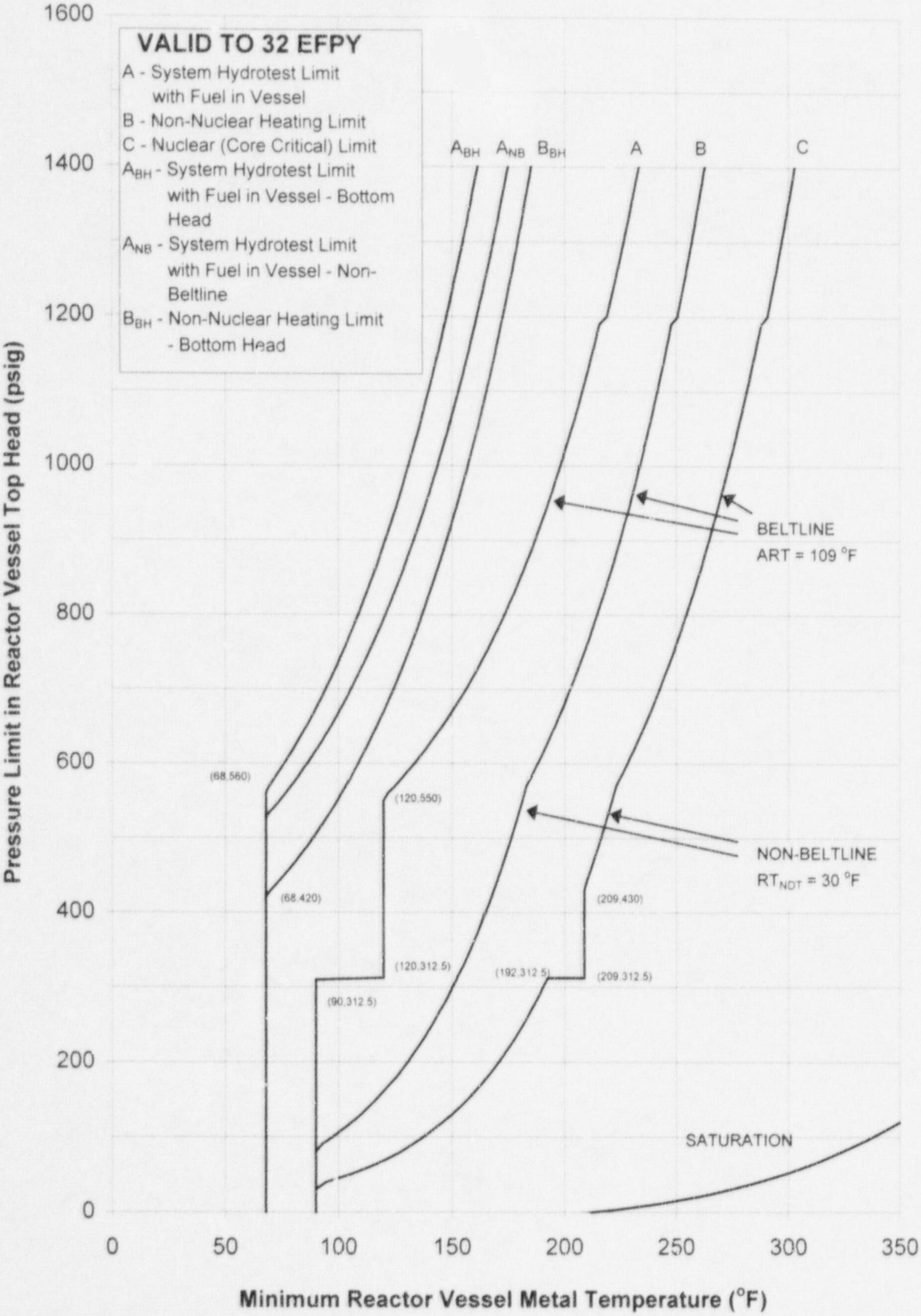

Figure 3.6-1 Part 2 Reactor Vessel Pressure-Temperature Limits Through 32 EFPY

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~~Figure 3.6-1 Part 3 Reactor Vessel Pressure - Temperature Limits Through 16 EFPY~~

ERRATA AND ADDENDA TO GE REPORT GE-NE-B1100732-01, REVISION 1

(JPTS-99-003)

New York Power Authority

JAMES A. FITZPATRICK NUCLEAR POWER PLANT
Docket No. 50-333
DPR-59

BJB-9907
6/17/99

ATTACHMENT TO BJB-9907

ERRATA AND ADDENDA SHEET TO GE-NE-B1100732-01, REVISION 1

NEW YORK POWER AUTHORITY
DOCUMENT REVIEW STATUS
STATUS NO: 1 ☒ ACCEPTED; 2 ☐ ACCEPTED AS NOTED RESUBMITTAL NOT REQUIRED; 3 ☐ ACCEPTED AS NOTED RESUBMITTAL REQUIRED; 4 ☐ NOT ACCEPTED
Permission to proceed does not constitute acceptance or approval of desig details, calculations, analysis, test methods or materials developed or selecte by the supplier and does not relieve supplier from full compliance wit contractual negotiations.
REVIEWED BY: G L Botta TITLE: Supv Engr
DATE: 6/22/9

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Approved by: T A Caine

T. A. Caine, Manager
Structural Assessment and Mitigation

ERRATA AND ADDENDA SHEET TO GE-NE-B1100732-01, REVISION 1

Page & Paragraph Number	Before Change	After Change
Page 73, 3rd paragraph	The 90°F limit applies when the head is on and tensioned, and also, when the head is off. (When fuel has been removed ...	The 90°F limit applies when the head is on and tensioned. The limiting vessel temperature is equal to the limiting RT_{NDT} of the vessel materials for two conditions: 1) When fuel is in the vessel and the head is off, or 2) When fuel has been removed from the vessel, the head is tensioned, and the pressure is below 20 psig.
Pages 70 through 75 Section 8 Sections 8.1 - 8.2.2	No non-beltline (i.e., upper vessel - feedwater) curve for the pressure test (Curve A) condition was provided.	Appendix C was added to the report to include a non-beltline (i.e., upper vessel - feedwater) curve for the pressure test (Curve A) condition.

APPENDIX C

FITZPATRICK

P-T CURVE CALCULATION METHOD

FOR THE NON-BELTLINE (UPPER VESSEL - FEEDWATER)

CURVE A

P-T CURVE CALCULATION METHOD

C.1 BACKGROUND

This appendix is an addenda to Section 8.2 of the report GE-NE-B1100732-01, Revision 1 [1]. The purpose of the addenda is to add a pressure-temperature (P-T) curve to be used specifically for the non-beltline (i.e., upper vessel - feedwater nozzle) during the hydrostatic pressure test and leak test operating conditions. Note that this curve is not to be used for the flange region.

There are four vessel regions defined in the thermal cycle diagram [2] that should be monitored against the P-T curve operating limits:

- Closure flange region (Region A)
- Core beltline region (Region B)
- Upper vessel (Regions A & B)
- Lower vessel (Regions B & C)

The closure flange region includes the bolts, top head flange, vessel flange, and adjacent plates and welds. The P-T curve methodology for the closure flange region is described in Section 8.3 of GE-NE-B1100732-01, Revision 1 [1]. The core beltline is the vessel location adjacent to the active fuel, such that the neutron fluence is sufficient to cause a significant shift of RT_{NDT}. The P-T curve methodology for the core beltline region is described in Sections 8.2.5 through 8.2.9 of GE-NE-B1100732-01, Revision 1 [1]. The remaining portion of the vessel (i.e., upper vessel , lower vessel) includes shells, components like the nozzles, the support skirt, and stabilizer brackets; these regions will be called the non-beltline region. The P-T curve methodology for the lower vessel region is described in Section 8.2.2 of GE-NE-B1100732-01, Revision 1 [1]. The P-T curve methodology for Core Not Critical Heat-up/Cool-down curve for the upper vessel region is described in Section 8.2.3 through 8.2.4 of GE-NE-B1100732-01, Revision 1 [1].

Under certain conditions, the minimum non-beltline (i.e., upper vessel - feedwater nozzle) temperature can be significantly cooler than the beltline or closure flange region. These conditions can occur when Reactor Water Clean-Up is used to make up to the vessel through the feedwater (FW) nozzle. To account for these circumstances, individual temperature limits for the non-beltline (i.e., upper vessel - feedwater nozzle) were established. The P-T curve methodology for pressure test curve for the non-beltline (i.e., upper vessel - feedwater nozzle) region is described in the following sections of this appendix.

The P-T curves for the heat-up and cool-down operating condition apply for both the 1/4T and 3/4T locations. When combining pressure and thermal stresses, it is usually necessary to evaluate stresses at the 1/4T location (inside surface flaw) and the 3/4T location (outside surface flaw). This is because the thermal gradient tensile stress of interest is in the inner wall during cool-down and is in the outer wall during heat-up. However, as a conservative simplification, the thermal gradient stress at the 1/4T is assumed to be tensile for both heat-up and cool-down. This results in the approach of applying the maximum tensile stress at the 1/4T location. This approach is conservative because the maximum stress is used regardless of flaw location.

C.2 NON-BELTLINE REGIONS

A discussion regarding the general methodology for the non-beltline regions is described in Section 8.2.1 of GE-NE-B1100732-01, Revision 1 [1].

As described in Section 8.2.1 plots were developed for the limiting BWR/4 components; the feedwater nozzle (FW) and the control rod drive (CRD) penetration (bottom head). All other components in the non-beltline regions are categorized under one of these two components as described in Tables C-1 and C-2 below:

TABLE C-1 APPLICABLE BWR/4 DISCONTINUITY COMPONENTS FOR USE WITH UPPER VESSEL - FEEDWATER CURVES A&B

Discontinuity Identification
FW Nozzle
CRD HYD System Return
Core Spray Nozzle
Recirculation Inlet Nozzle
Steam Outlet Nozzle
Water Level Instrumentation Nozzle
Main Closure Flange
Support Skirt
Stabilizer Brackets
Vibration Instrumentation Nozzle
Core ΔP and Liquid Control Nozzle
Steam Water Interface
Jet Pump Instrumentation Nozzle
Shell
CRD and Bottom Head (B only)
Top Head Nozzles (B only)
Recirculation Outlet Nozzle (B only)

TABLE C-2 APPLICABLE BWR/4 DISCONTINUITY COMPONENTS FOR USE WITH BOTTOM HEAD/CRD CURVES A & B

Discontinuity Identification
CRD and Bottom Head
Top Head Nozzle
Recirculation Outlet Nozzle

The P-T curves for the non-beltline region were conservatively developed for a large BWR/6 (nominal inside diameter of 251 inches). The analysis is considered appropriate for FitzPatrick as the plant specific geometric values are bounded by the generic analysis for a large BWR/6, as determined from equations C-1 and C-2. The generic value was adapted to the conditions at FitzPatrick by using plant specific RT_{NDT} values for the reactor pressure vessel (RPV). The presence of nozzles and control rod (CRD) penetration holes of the non-beltline (i.e., upper vessel and bottom head, respectively), has made the analysis different from a shell analysis such as the beltline. This was the result of the stress concentrations and higher thermal stresses for certain transient conditions experienced by the non-beltline region (i.e., upper vessel and the bottom head).

The non-beltline curves are shifted based on the most limiting initial RT_{NDT} values for the appropriate non-beltline components; the initial RT_{NDT} values are listed in Table 3-2 of GE-NE-B1100732-01, Revision 1 [1]. The individual non-beltline (i.e., upper vessel - feedwater nozzle) curve is based on the non-beltline feedwater nozzle curve described in the next section.

C.2.1 Pressure Test - Non-Beltline Curve A (Using Upper Vessel - Feedwater Nozzle Region)

CBI Nuclear (CBIN) modeled the 251 inch BWR/6 feedwater nozzles [3] to compute local stresses for determination of the stress intensity factor, K_I. The result of that computation was K_I = 143.1 ksi-in$^{1/2}$ for an applied pressure of 1563 psig preservice hydrotest pressure. The computed value of ($T-RT_{NDT}$) was 154°F. The respective flaw depth and orientation used in this calculation is perpendicular to the maximum stress (hoop) at a depth of 1/4T through the corner thickness.

To evaluate the CBIN result, K_I is calculated for the upper vessel nominal stress, PR/t, according to the methods in ASME Code Appendix G (Section III). The result is compared to that determined by CBIN in order to quantify the K magnification associated with the stress concentration created by the feedwater nozzles.

A calculation of K_I is shown below using the BWR/6, 251-inch dimensions:

Vessel Radius, R	126.7 inches
Vessel Thickness, t	6.5 inches
Vessel Pressure	1563 psig

Pressure stress: $\sigma = PR/t$ = 1563 psig * 126.7 inches/(6.5 inches)

= 30466 psi

The factor F (a/Rn) from Figure A5-1 of WRC-175 is 1.6 where :

$$a = \text{lesser of } 1/4 \text{ Tn or } 1/4 \text{ Tv}$$

$$Tn = 7\ 1/8 \text{ inches}$$

$$Tv = 6\ 1/2 \text{ inches}$$

$$Rn = \text{apparent radius of nozzle} = Ri + 0.29\ Rc$$

$$Ri = \text{actual inner radius of nozzle} = 6 \text{ inches}$$

$$Rc = \text{nozzle radius (nozzle corner radius)} = 4.0 \text{ inches}$$

Thus, $a/Rn = 1.63/6.94 = 0.23$ and the ratio of K_I around the feedwater nozzle to the membrane stress * $(\pi a)^{1/2}$ at places with no geometric discontinuity is 1.6.

Including the safety factor of 1.3, the stress intensity factor, K_I, is $1.3\ \sigma\ (\pi a)^{1/2} * F(a/Rn)$:

$$\text{Nominal } K_I = 1.3 * 30.466 * (\pi * 1.63)^{1/2} * 1.6 = 143 \text{ ksi-in}^{1/2}$$

The method to solve for $(T-RT_{NDT})$ for a specific K_I is based on the curve in Figure G-2210-1 in ASME Appendix G:

$$(T-RT_{NDT}) = \ln [(K_I - 26.78) / 1.223] / 0.0145 - 160$$

$$(T-RT_{NDT}) = \ln [(143 - 26.78) / 1.223] / 0.0145 - 160$$

$$(T-RT_{NDT}) = 154°F$$

The generic pressure test P-T curve was generated by scaling 143 ksi-in$^{1/2}$ by the nominal pressures and calculating the associated $(T-RT_{NDT})$:

Pressure Test Feedwater Nozzle K_I and $(T - RT_{NDT})$ as a Function of Pressure

Nominal Pressure (psig)	K_I (ksi-in$^{1/2}$)	$(T-RT_{NDT})$ (°F)
1563	143	154
1400	128	145
1200	110	131
1000	92	114
800	73	91
600	55	56
400	37	-16

The P-T curve is dependent on the K_I value calculated, which is proportional to the stress and the crack depth according to the relationship:

$$K_I \propto \sigma (\pi a)^{1/2}$$

The stress is proportional to R/t and, for the P-T curves, crack depth, a, is t/4. Thus, K_I is proportional to $R/(t)^{1/2}$.

The generic curve value of $R/(t)^{1/2}$, based on the BWR/6, 251-inch feedwater nozzle dimensions is:

$$\text{Generic:} \quad R/(t)^{1/2} = \quad 126.7 / (6.5)^{1/2} = \quad 49.7 \text{ inch}^{1/2}, \qquad \text{(C-1)}$$

where t is the nominal vessel thickness. The FitzPatrick specific vessel shell dimensions applicable to the feedwater nozzle location are R = 110 inches and t = 5.9 inches nominal.

$$\text{FitzPatrick specific:} \quad R/(t)^{1/2} = 110 / (5.9)^{1/2} = 45 \text{ inch}^{1/2} \qquad \text{(C-2)}$$

Since the generic value of $R/(t)^{1/2}$ is greater than that for FitzPatrick, the generic P-T curve is conservative when applied to the FitzPatrick feedwater nozzle.

The highest RT_{NDT} for the feedwater region component (nozzle #N2, the Recirculation Inlet Nozzle) at FitzPatrick is 30°F as shown in Table 3-2 of GE-NE-B1100732-01, Revision 1 [1]. The generic pressure test P-T curve is applied to the FitzPatrick feedwater nozzle curve by shifting the P vs. ($T-RT_{NDT}$) values above to reflect the RT_{NDT} value of 30°F. This non-beltline (i.e., upper vessel - feedwater nozzle) P-T curve is tabulated in Table C-3. The only difference between Table C-3 and Table 8-1 is the addition of the non-beltline (i.e., upper vessel - feedwater nozzle) curve tabulation. Note that this curve does not apply to the flange region. Since non-beltline curves are not influenced by irradiation, the curve is applicable for any EFPY.

TABLE C-3. FitzPatrick P-T Curve Values for 32 EFPY

Replaces Table 8-1 of Report GE-NE-B11-00732-01, Revision 1

Required Temperatures at 100°F/hr for Curves B & C and 20°F/hr for Curve A

PRESSURE	BOTTOM HEAD CURVE A	NON-BELTLINE CURVE A	RPV & 32 EFPY BELTLINE CURVE A	BOTTOM HEAD CURVE B	RPV & 32 EFPY BELTLINE CURVE B	RPV & 32 EFPY BELTLINE CURVE C
(PSIG)	(°F)	(°F)	(°F)	(°F)	(°F)	(°F)
0	68.0	68.0	90.0	68.0	90.0	90.0
10	68.0	68.0	90.0	68.0	90.0	90.0
20	68.0	68.0	90.0	68.0	90.0	90.0
30	68.0	68.0	90.0	68.0	90.0	90.0
40	68.0	68.0	90.0	68.0	90.0	94.5
50	68.0	68.0	90.0	68.0	90.0	105.2
60	68.0	68.0	90.0	68.0	90.0	113.9
70	68.0	68.0	90.0	68.0	90.0	121.1
80	68.0	68.0	90.0	68.0	90.0	127.4
90	68.0	68.0	90.0	68.0	92.7	132.7
100	68.0	68.0	90.0	68.0	97.5	137.5
110	68.0	68.0	90.0	68.0	101.9	141.9
120	68.0	68.0	90.0	68.0	106.1	146.1
130	68.0	68.0	90.0	68.0	110.1	150.1
140	68.0	68.0	90.0	68.0	113.6	153.6
150	68.0	68.0	90.0	68.0	116.8	156.8
160	68.0	68.0	90.0	68.0	119.8	159.8
170	68.0	68.0	90.0	68.0	122.8	162.8
180	68.0	68.0	90.0	68.0	125.6	165.6
190	68.0	68.0	90.0	68.0	128.2	168.2
200	68.0	68.0	90.0	68.0	130.6	170.6
210	68.0	68.0	90.0	68.0	132.9	172.9
220	68.0	68.0	90.0	68.0	135.2	175.2
230	68.0	68.0	90.0	68.0	137.4	177.4
240	68.0	68.0	90.0	68.0	139.4	179.4
250	68.0	68.0	90.0	68.0	141.4	181.4
260	68.0	68.0	90.0	68.0	143.3	183.3
270	68.0	68.0	90.0	68.0	145.1	185.1
280	68.0	68.0	90.0	68.0	147.0	187.0
290	68.0	68.0	90.0	68.0	148.7	188.7
300	68.0	68.0	90.0	68.0	150.3	190.3
310	68.0	68.0	90.0	68.0	152.0	192.0
312.5	68.0	68.0	90.0	68.0	152.3	192.3
312.5	68.0	68.0	120.0	68.0	152.3	208.7
320	68.0	68.0	120.0	68.0	153.5	208.7
330	68.0	68.0	120.0	68.0	155.1	208.7

TABLE C-3. FitzPatrick P-T Curve Values for 32 EFPY

Replaces Table 8-1 of Report GE-NE-B11-00732-01, Revision 1

Required Temperatures at 100°F/hr for Curves B & C and 20°F/hr for Curve A

PRESSURE	BOTTOM HEAD CURVE A	NON-BELTLINE CURVE A	RPV & 32 EFPY BELTLINE CURVE A	BOTTOM HEAD CURVE B	RPV & 32 EFPY BELTLINE CURVE B	RPV & 32 EFPY BELTLINE CURVE C
(PSIG)	(°F)	(°F)	(°F)	(°F)	(°F)	(°F)
340	68.0	68.0	120.0	68.0	156.6	208.7
350	68.0	68.0	120.0	68.0	158.0	208.7
360	68.0	68.0	120.0	68.0	159.4	208.7
370	68.0	68.0	120.0	68.0	160.8	208.7
380	68.0	68.0	120.0	68.0	162.1	208.7
390	68.0	68.0	120.0	68.0	163.4	208.7
400	68.0	68.0	120.0	68.0	164.7	208.7
410	68.0	68.0	120.0	68.0	166.0	208.7
420	68.0	68.0	120.0	68.0	167.2	208.7
430	68.0	68.0	120.0	70.3	168.4	208.7
440	68.0	68.0	120.0	73.2	169.6	209.6
450	68.0	68.0	120.0	76.1	170.7	210.7
460	68.0	68.0	120.0	78.8	171.8	211.8
470	68.0	68.0	120.0	81.5	172.9	212.9
480	68.0	68.0	120.0	84.0	174.0	214.0
490	68.0	68.0	120.0	86.5	175.1	215.1
500	68.0	68.0	120.0	88.8	176.1	216.1
510	68.0	68.0	120.0	91.1	177.1	217.1
520	68.0	68.0	120.0	93.3	178.1	218.1
530	68.0	68.5	120.0	95.5	179.1	219.1
540	68.0	71.3	120.0	97.6	180.1	220.1
550	68.0	74.1	120.0	99.6	181.1	221.1
560	68.0	76.7	122.0	101.5	182.0	222.0
570	69.5	79.2	125.1	103.5	182.9	222.9
580	71.8	81.7	128.0	105.3	184.6	224.6
590	74.0	84.0	130.8	107.1	186.3	226.3
600	76.1	86.3	133.6	108.9	187.9	227.9
610	78.2	88.5	136.2	110.6	189.5	229.5
620	80.2	90.6	138.7	112.3	191.1	231.1
630	82.1	92.7	141.1	113.9	192.6	232.6
640	84.0	94.7	143.5	115.5	194.1	234.1
650	85.9	96.7	145.8	117.1	195.6	235.6
660	87.7	98.6	148.0	118.6	197.0	237.0
670	89.4	100.4	150.1	120.1	198.4	238.4
680	91.1	102.2	152.2	121.6	199.8	239.8
690	92.8	104.0	154.2	123.0	201.1	241.1

TABLE C-3. FitzPatrick P-T Curve Values for 32 EFPY

Replaces Table 8-1 of Report GE-NE-B11-00732-01, Revision 1

Required Temperatures at 100°F/hr for Curves B & C and 20°F/hr for Curve A

PRESSURE (PSIG)	BOTTOM HEAD CURVE A (°F)	NON-BELTLINE CURVE A (°F)	RPV & 32 EFPY BELTLINE CURVE A (°F)	BOTTOM HEAD CURVE B (°F)	RPV & 32 EFPY BELTLINE CURVE B (°F)	RPV & 32 EFPY BELTLINE CURVE C (°F)
700	94.4	105.7	156.1	124.4	202.4	242.4
710	96.0	107.4	158.0	125.8	203.7	243.7
720	97.6	109.0	159.9	127.1	205.0	245.0
730	99.1	110.6	161.7	128.4	206.3	246.3
740	100.6	112.2	163.4	129.7	207.5	247.5
750	102.0	113.7	165.1	131.0	208.7	248.7
760	103.5	115.2	166.8	132.3	209.9	249.9
770	104.8	116.7	168.4	133.5	211.1	251.1
780	106.2	118.1	170.0	134.7	212.2	252.2
790	107.6	119.5	171.6	135.9	213.3	253.3
800	108.9	120.9	173.1	137.0	214.4	254.4
810	110.2	122.2	174.6	138.2	215.5	255.5
820	111.4	123.5	176.0	139.3	216.6	256.6
830	112.7	124.8	177.5	140.4	217.7	257.7
840	113.9	126.1	178.9	141.5	218.7	258.7
850	115.1	127.3	180.2	142.6	219.7	259.7
860	116.3	128.6	181.6	143.6	220.8	260.8
870	117.5	129.8	182.9	144.7	221.8	261.8
880	118.6	130.9	184.2	145.7	222.7	262.7
890	119.7	132.1	185.5	146.7	223.7	263.7
900	120.8	133.3	186.7	147.7	224.7	264.7
910	121.9	134.4	187.9	148.7	225.6	265.6
920	123.0	135.5	189.1	149.7	226.5	266.5
930	124.0	136.6	190.3	150.6	227.5	267.5
940	125.1	137.7	191.5	151.6	228.4	268.4
950	126.1	138.7	192.6	152.5	229.3	269.3
960	127.1	139.8	193.7	153.4	230.1	270.1
970	128.1	140.8	194.9	154.3	231.0	271.0
980	129.1	141.8	195.9	155.2	231.9	271.9
990	130.0	142.8	197.0	156.1	232.7	272.7
1000	131.0	143.8	198.1	157.0	233.6	273.6
1010	131.9	144.7	199.1	157.8	234.4	274.4
1020	132.9	145.7	200.1	158.7	235.2	275.2
1030	133.8	146.6	201.1	159.5	236.0	276.0
1040	134.7	147.6	202.1	160.4	236.8	276.8
1050	135.6	148.5	203.1	161.2	237.6	277.6

TABLE C-3. FitzPatrick P-T Curve Values for 32 EFPY

Replaces Table 8-1 of Report GE-NE-B11-00732-01, Revision 1

Required Temperatures at 100°F/hr for Curves B & C and 20°F/hr for Curve A

PRESSURE	BOTTOM HEAD CURVE A	NON-BELTLINE CURVE A	RPV & 32 EFPY BELTLINE CURVE A	BOTTOM HEAD CURVE B	RPV & 32 EFPY BELTLINE CURVE B	RPV & 32 EFPY BELTLINE CURVE C
(PSIG)	(°F)	(°F)	(°F)	(°F)	(°F)	(°F)
1060	136.5	149.4	204.1	162.0	238.4	278.4
1070	137.3	150.3	205.0	162.8	239.2	279.2
1080	138.2	151.2	206.0	163.6	239.9	279.9
1090	139.0	152.0	206.9	164.4	240.7	280.7
1100	139.9	152.9	207.8	165.1	241.4	281.4
1110	140.7	153.7	208.7	165.9	242.2	282.2
1120	141.5	154.6	209.6	166.7	242.9	282.9
1130	142.3	155.4	210.5	167.4	243.6	283.6
1140	143.1	156.2	211.4	168.1	244.4	284.4
1150	143.9	157.0	212.2	168.9	245.1	285.1
1160	144.7	157.8	213.1	169.6	245.8	285.8
1170	145.5	158.6	213.9	170.3	246.5	286.5
1180	146.2	159.4	214.7	171.0	247.2	287.2
1190	147.0	160.2	215.6	171.7	247.8	287.8
1200	147.7	160.9	218.8	172.4	250.6	290.6
1210	148.5	161.7	219.6	173.1	251.2	291.2
1220	149.2	162.4	220.4	173.8	251.9	291.9
1230	149.9	163.2	221.1	174.5	252.5	292.5
1240	150.6	163.9	221.9	175.1	253.2	293.2
1250	151.3	164.6	222.7	175.8	253.8	293.8
1260	152.0	165.4	223.4	176.5	254.5	294.5
1270	152.7	166.1	224.2	177.1	255.1	295.1
1280	153.4	166.8	224.9	177.8	255.7	295.7
1290	154.1	167.5	225.6	178.4	256.3	296.3
1300	154.8	168.1	226.3	179.0	256.9	296.9
1310	155.4	168.8	227.0	179.6	257.5	297.5
1320	156.1	169.5	227.7	180.3	258.2	298.2
1330	156.8	170.2	228.4	180.9	258.7	298.7
1340	157.4	170.8	229.1	181.5	259.3	299.3
1350	158.1	171.5	229.8	182.1	259.9	299.9
1360	158.7	172.1	230.5	182.7	260.5	300.5
1370	159.3	172.8	231.1	183.3	261.1	301.1
1380	159.9	173.4	231.8	183.9	261.7	301.7
1390	160.6	174.0	232.5	184.5	262.2	302.2
1400	161.2	174.7	233.1	185.0	262.8	302.8

C.6 REFERENCES

[1] T. J. Griesbach, "Plant FitzPatrick RPV Surveillance Materials Testing and Analysis of 120°F Capsule at 13.4 EFPY," GE-NE, San Jose, CA, February 1998, (GE-NE-B1100732-01, Rev. 1).

[2] GE Drawing #729E762, Revision 0, "Reactor Thermal Cycles," GE-APED, San Jose, CA.

[3] Appendix C1 "Analysis of the Nozzle to Shell Junction Region for 10" Welded Thermal Sleeve Nozzles," CBI Nuclear Company, (GE VPF# 3521-415-6).