Entergy

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10CFR50.90

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ENGC Ltr. 2.00.080

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D.C. 20555-0001

Docket No. 50-293
License No. DPR-35

REQUEST FOR TECHNICAL SPECIFICATION CHANGE
CONCERNING PRESSURE-TEMPERATURE LIMIT CURVES OF FIGURES 3.6.1, 3.6.2, AND 3.6.3

Entergy Nuclear Generation Company (ENGC)-Pilgrim requests NRC review and approval for a change to the pressure-temperature limit curves of Figures 3.6.1, 3.6.2, and 3.6.3 of Pilgrim's Technical Specifications to cover operation between 20, 32 and 48 Effective Full Power Years. Also included are changes to the Bases section. The attached "No Significant Hazards Considerations" evaluation is provided with the proposed change.

The current pressure-temperature limit curves of Figures 3.6.1, 3.6.2, and 3.6.3 were incorporated into Pilgrim's Technical Specifications by Amendment No. 140, effective January 29, 1992. This proposed change was developed in accordance with Appendices G and H of 10 CFR Part 50; the ASTM Standards and the ASME Code (including Code Cases N-588 and N-640), which are referenced in Appendices G and H, 10 CFR 50.36(c)(2); Regulatory Guide 1.99, Revision 2, *Radiation Embrittlement of Reactor Vessel Materials* and Generic Letter 88-11.

- **ASME Code Case N-588, "Alternative to Reference Flaw Orientation of Appendix G for Circumferential Welds in Reactor Vessels, Section XI, Division 1."** This Code Case allows the use of alternative procedures for defining the orientation of postulated flaws in circumferential welds and for calculating the applied stress intensity factors of axial and circumferential flaws. Code Case N-588 was approved for use by the appropriate ASME Boiler & Pressure Vessel Section XI committee on December 12, 1997.

- **ASME Code Case N-640, "Alternative Reference Fracture Toughness for Development of P-T Curves for ASME Section XI, Division 1."** This Code Case provides an alternate method for determining the fracture toughness of reactor vessel materials for use in determining P-T Limits. Code Case N-640 was approved for use by the appropriate ASME Boiler & Pressure Vessel Section XI committee on February 26, 1999.

Code Cases N-588 and N-640 are pending NRC approval.

Data from the first capsule pull (4.17 EFPY) indicates Pilgrim's existing pressure-temperature curves (Technical Specification Figures 3.6-1, 3.6-2, and 3.6-3) represent unnecessarily conservative values. This proposed amendment reflects new information concerning Pilgrim's reactor vessel material and

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fabrication provided by the reactor vessel manufacturer (Combustion Engineering). These proposed changes provide more operating margin in the pressure-temperature limit curves for in-service leakage and hydrostatic pressure testing, non-nuclear heatup and cooldown, and criticality. This proposed change offers potential radiation dose savings by increasing the effectiveness of inspectors in the containment at lower ambient temperature and potential outage schedule savings. Operating Pilgrim consistent with the regime defined by these proposed curves provides assurance of reactor vessel integrity for the remainder of plant life while also providing greater operational flexibility.

Please contact P.M. Kahler at (508) 830-7939 if you require further information on this issue.

Sincerely,

Mike Bellamy

Commonwealth of Massachusetts)
County of Plymouth)

Then personally appeared before me, Mike Bellamy, who being duly sworn, did state that he is Site Vice President, Entergy Nuclear Generation Company and that he is duly authorized to execute and file the submittal contained herein in the name and on behalf of Entergy Nuclear Generation Company and that under the penalty of perjury the foregoing is true and correct.

My commission expires: September 20, 2002 Peter M Kahler
DATE NOTARY PUBLIC

Attachments:

1) Narrative on Proposed Change and "No Significant Hazards Consideration"
2) Proposed Changed Pilgrim Technical Specification Pages 3/4.6-14, 3/4.6-15, 3/4.6-16, B3/4.6-2, and B3/4.6-3
3) Marked-up Current Pilgrim Technical Specification Pages 3/4.6-14, 3/4.6-15, 3/4.6-16, B3/4.6-2, and B3/4.6-3

RMB/PMK/

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Description of Proposed Change

The proposed amendment revises the pressure-temperature limit curves of Figures 3.6.1, 3.6.2, and 3.6.3 of Pilgrim's Technical Specifications to cover operation between 20, 32, and 48 effective full power years (EFPY). Also included are changes to the Bases section.

Reason for Change

Revision of the pressure-temperature limit curves of Figures 3.6.1, 3.6.2, and 3.6.3 reflects new information concerning Pilgrim's reactor vessel material and fabrication provided by the reactor vessel manufacturer. The revision also incorporates new methodology available for pressure-temperature curve development, including application of ASME Code Cases N-588 and N-640. These proposed changes provide more operating margin in the pressure-temperature limit curves for in-service leakage and hydrostatic pressure testing, non-nuclear heatup and cooldown, and criticality. This proposed change offers potential radiation dose savings by increasing the effectiveness of inspectors in the containment at lower ambient temperature and potential outage schedule savings. Operating Pilgrim consistent with the regime defined by these proposed curves provides assurance of reactor vessel integrity for the remainder of plant life while also providing greater operational flexibility.

Safety Evaluation

Entergy Nuclear Generation Company (ENGC) requests approval of the prposed changes to the Pilgrim Nuclear Power Station Technical Specifications. The requested changes would revise the pressure-temperature limits in Pilgrim Technical Specifications, Figures 3.6-1, 3.6-2, and 3.6-3. The proposed pressure-temperature limits are requested for 20, 32 and 48 effective full power years (EFPY). Pilgrim's current EFPY is about 16.3. The material properties associated with the proposed pressure-temperature limits were developed using Regulatory Guide (RG) 1.99, Revision 2. Generic Letter 88-11, "NRC Position on Radiation Embrittlement of Reactor Vessel Materials and Its Effect on Plant Operations," recommends the use of RG 1.99, Rev. 2, for calculating pressure-temperature limits. The proposed changes were developed in accordance with the following NRC regulations and guidance: Appendices G and H of 10 CFR Part 50; the ASTM Standards and the ASME Code Section XI (including Code Cases N-588 and N-640), in Appendices G and H, 10 CFR 50.36(c)(2); RG 1.99, Rev. 2; and Generic Letter 88-11.

Appendix G of 10 CFR Part 50 specifies fracture toughness and testing requirements for reactor vessel materials in accordance with the ASME Code. These tests define the extent of vessel embrittlement at the time of capsule withdrawal in terms of the increase reference temperature. Appendix G also requires prediction of the effects of neutron irradiation on vessel embrittlement by calculating the adjusted reference temperature (ART) and Charpy upper shelf energy (USE). Generic Letter 88-11 requested that the methods provided in RG 1.99, Rev. 2, be used to predict the effect of neutron irradiation on reactor vessel materials. This guide defines the ART as the sum of unirradiated reference temperature, the increase in reference temperature resulting from neutron irradiation, and a margin to account for uncertainties in the prediction method.

Pilgrim evaluated the effect of neutron irradiation embrittlement on each beltline material in Pilgrim's reactor vessel. The amount of irradiation embrittlement was calculated in accordance with RG 1.99, Rev. 2. Pilgrim determined that the material with the highest ART at 32 EFPY was the lower intermediate shell axial weld (1-338 A, B, and C) with 0.219% copper (Cu), 0.996% nickel (Ni), and an initial RT_{ndt} of –48 degrees F.

Pilgrim removed one surveillance capsule at 4.17 EFPY. The test results from this capsule were published in Southwest Research Institute Report SWRI 02-5951. All surveillance capsules contained Charpy impact specimens and tensile specimens made from base metal, weld metal, and heat affected zone (HAZ) metal.

For the limiting beltline material, weld 1-338 A, B, and C, Pilgrim calculated the ART at 32 EFPY to be 105 degrees F at 1/4T (T=reactor vessel beltline thickness) and 124 degrees F for 1/4T at 48 EFPY. Pilgrim based these adjusted reference temperature shifts on a calculated neutron fluence of 9.95E17 n/cm^2 at 1/4T and 1.48E18 n/cm^2 at 1/4T for 48 EFPY. The ART was determined by Section 1 of RG 1.99, Rev. 2. The 2σ margin of 56 degrees F was used for the welds because only one capsule has been removed from the Pilgrim reactor pressure vessel.

Using the ART of 84, 105, and 124 degrees F for the 20, 32 and 48 EFPY conditions, respectively, in the equations of Reg. Guide 1.99, Rev. 2, for the most limiting component, weld 1-338 A, confirmed the proposed pressure-temperature limits for 20, 32 and 48 EFPY for heatup cooldown, and hydrotest meet the beltline material requirements in Appendix G of 10 CFR Part 50.

In addition to beltline materials, Appendix G of 10 CFR Part 50 also imposes pressure-temperature limits based on the reference temperature for the reactor vessel closure flange materials. Section IV.A.2 of Appendix G states that when the pressure exceeds 20% of the pre-service system hydrostatic test pressure, the temperature of the closure flange regions highly stressed by the bolt preload must exceed the reference temperature of the material in those regions by at least 120 degrees F for normal operation and by 90 degrees F for hydrostatic pressure tests and leak tests. Paragraph IV.A.3 of Appendix G states, "...an exception may be made for boiling water reactor vessels when water level is within the normal range for power operation and the pressure is less than 20 percent of the pre-service system hydrostatic test pressure. In this case the minimum permissible temperature is 60 degrees F (33 degrees C) above the reference temperature of the closure flange regions that are highly stressed by the bolt preload." Based on the flange reference temperature of 10 degrees F, Pilgrim has determined that imposing the proposed pressure-temperature limit of 70 degrees F through 48 EFPY satisfies Section IV.A.2 of Appendix G.

The reference temperature calculations for the reactor beltline materials as prescribed in RG 1.99, Rev. 2, are not applicable to the upper region materials since embrittlement due to irradiation is not significant in these regions. Pilgrim utilized bounding stresses for the bottom head region based on review of all design basis stresses due to internal pressure and all relevant thermal transients originally specified for the Pilgrim RPV. These stresses encompass large heatup and cooldown rates such as those associated with inadvertent cooldown events in the bottom head region. Therefore, the revised pressure-temperature curves are expected to cover all anticipated operating conditions, as defined in 10 CFR 50, Appendix A. Any events that exceed those considered in the development of the revised pressure-temperature curves are monitored via the Technical Specification pressure-temperature limit tracking procedures, coupled with ASME Code, Section XI, Appendix E evaluation procedures. The maximum stresses, stress intensity factors and pressure-temperature limits were calculated based on ASME Code, Section XI, Appendix G and 10 CFR 50, Appendix G, with appropriate modifications to allow for incorporation of ASME Code Cases N-588 and N-640. To safeguard the structural integrity of the reactor beltline materials, Pilgrim will continue to ensure the temperature readings from the temperature sensors are within the acceptable region of the beltline pressure-temperature limit.

No Significant Hazards Considerations

The proposed amendment revises the thermal and pressurization limit curves of Figures 3.6.1, 3.6.2, and 3.6.3 of Pilgrim's Technical Specifications to cover operation between 20 and 32 and 48 EFPY. Also included are changes to the Bases section.

As required by 10 CFR 50.91(a), Pilgrim has provided its analysis of the issue of no significant hazards consideration, which is presented below:

- **The proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.**

The proposed Technical Specification changes do not involve a significant increase in the probability or consequences of an accident previously evaluated. There are no physical changes to the plant being introduced by the proposed changes to the pressure-temperature curves. The proposed changes do not modify the reactor coolant pressure boundary, (i.e., there are no changes in operating pressure, materials or seismic loading). The proposed changes do not adversely affect the integrity of the reactor coolant pressure boundary such that its function in the control of radiological consequences is affected. The proposed pressure-temperature curves were generated in accordance with the fracture toughness requirements of 10 CFR Part 50, Appendix G, and American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code, Section XI, Appendix G, in conjunction with ASME Code Cases N-640 and N-588. The proposed pressure-temperature curves were established in compliance with the methodology used to calculate and predict effects of radiation on embrittlement of reactor vessel beltline materials. Usage of these procedures provides compliance with the intent of 10 CFR Part 50, Appendix G, and provides margins of safety that ensure failure of the reactor vessel will not occur.

The changes to the calculation methodology for the pressure-temperature limits based upon Code Cases N-640 and N-588 provide sufficient margin in the prevention of a non-ductile type fracture of the reactor pressure vessel. The code cases were developed using knowledge gained through years of industry experience. Pressure-temperature curves developed using the allowances of Code Cases N-640 and N-588 yield more operating margin. However, the experience gained in the areas of fracture toughness of materials and pre-existing undetected defects show that some of the previous assumptions used for the calculation of pressure-temperature limits are overly conservative. Therefore, providing the allowances of the subject code cases in developing the pressure-temperature limit curves will continue to provide protection against non-ductile-type fractures of the reactor pressure vessel.

Thus, the proposed pressure-temperature curves prohibit operation in regions where it is possible for brittle fracture of reactor vessel materials to occur. Hence, the primary coolant pressure boundary integrity will be maintained. Consequently, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

- **The proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.**

The proposed Technical Specification changes do not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed changes to the pressure-temperature curves were generated in accordance with the fracture toughness requirements of 10 CFR Part 50, Appendix G, and ASME B&PV Code, Section XI, Appendix G, in conjunction with ASME Code Cases N-588 and N-640. The proposed changes provide more operating margin in the pressure-temperature limit curves for in-service leakage and hydrostatic pressure testing, non-nuclear heatup and cooldown, and criticality. However, compliance with the proposed pressure-temperature curves will ensure conditions in which brittle fracture of primary coolant pressure boundary materials is possible will be avoided because such compliance with the proposed pressure-temperature curves provides sufficient protection against a non-ductile-type fracture of the reactor pressure vessel. No new modes of operation are introduced by the proposed changes. The proposed changes will not create any failure mode not bounded by previously evaluated accidents. Further, the proposed changes to the pressure-temperature curves do not affect any activities or equipment and are not assumed in any safety

analysis to initiate any accident sequence. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

- **The proposed changes do not involve a significant reduction in a margin of safety.**

The proposed Technical Specification changes do not involve a significant reduction in a margin of safety. The proposed changes reflect an update of the pressure-temperature curves. The revised curves are based on the latest NRC and ASME guidance.

The revised pressure-temperature limits have been developed using the K_{Ic} fracture toughness curve shown in ASME B&PV Code Section XI, Appendix A, Figure A-2000-1, in lieu of the K_{Ia} fracture toughness curve shown in ASME B&PV Code Section XI, Appendix G, Figure G-2010-1, as the lower bound fracture toughness. The other margins involved with the ASME B&PV Code, Section XI, Section G process of determining pressure-temperature limit curves remain unchanged.

The controlled heatup and cooldown process limits the rate at which stress is developed in the RPV wall to rates that are more appropriate for static initiation fracture toughness. Use of the K_{Ic} curve in determining the lower bound fracture toughness in the development of pressure-temperature operating limits is technically justified because this curve implements the static initiation fracture toughness; whereas, the K_{Ia} curve is based on lower bound crack arrest fracture toughness which is ultraconservative.

As mentioned previously, the revised pressure-temperature curves provide more operating margin and, thus, more operational flexibility than the current pressure-temperature curves. However, industry experience since the inception of the pressure-temperature limits in 1974 confirms that some of the original methodologies used to develop pressure-temperature curves are overly conservative. Accordingly, ASME Code Cases N-640 and N-588 take advantage of the acquired knowledge by establishing more realistic methodologies for the development of pressure-temperature curves. Therefore, operational flexibility is gained and an acceptable margin of safety to reactor pressure vessel non-ductile type fracture is maintained.

These revised pressure-temperature limits, although less restrictive than the current limits, are established in accordance with current regulations and the latest ASME Code information. Because operation will be within these limits, the reactor vessel materials will continue to behave in a non-brittle manner, thereby preserving the original safety design bases. No plant safety limits, set points, or design parameters are adversely affected by the proposed changes. Therefore, the proposed changes do not involve a significant reduction in a margin of safety.

These changes have been reviewed and recommended for approval by the Operations Review Committee, and reviewed by the Nuclear Safety Review and Audit Committee.

Environmental Consideration

The amendment changes a requirement with respect to use of a facility component located within the restricted area as defined in 10 CFR Part 20. Pilgrim has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. Pilgrim also finds that the proposed amendment involves no significant hazards consideration. Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Hence, pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

Schedule of Change

This change will be implemented within 30 days following Pilgrim's receipt of its approval by the Commission.

ATTACHMENT 2 TO PILGRIM LETTER

Proposed Changed Pilgrim
Technical Specification Figures 3.6.1, 3.6.2, 3.6.3 and Bases

PILGRIM REACTOR VESSEL PRESSURE-TEMPERATURE LIMITS
HYDROSTATIC AND LEAK TESTS

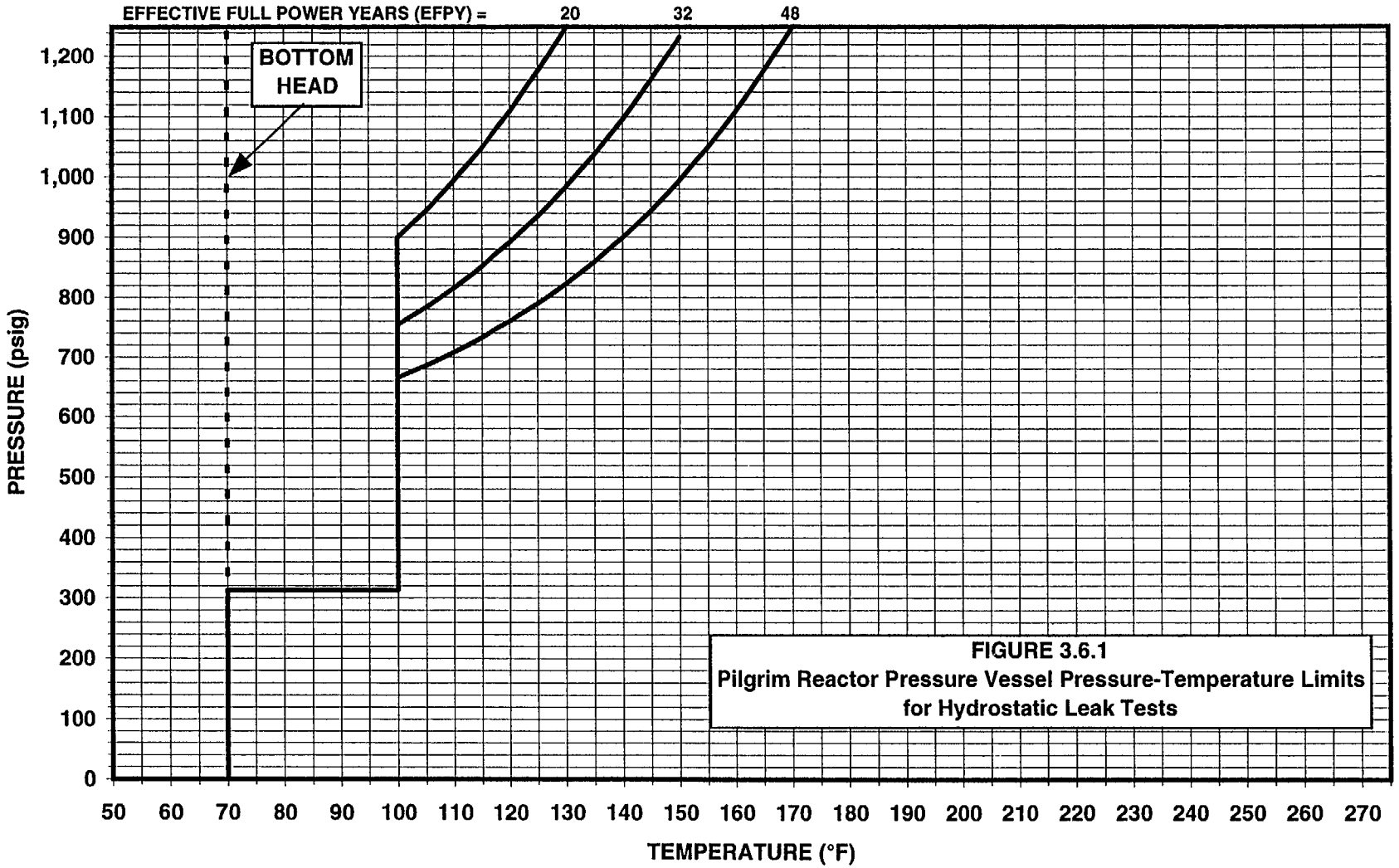

FIGURE 3.6.1
Pilgrim Reactor Pressure Vessel Pressure-Temperature Limits for Hydrostatic Leak Tests

PILGRIM REACTOR VESSEL PRESSURE-TEMPERATURE LIMITS
SUBCRITICAL HEATUP AND COOLDOWN

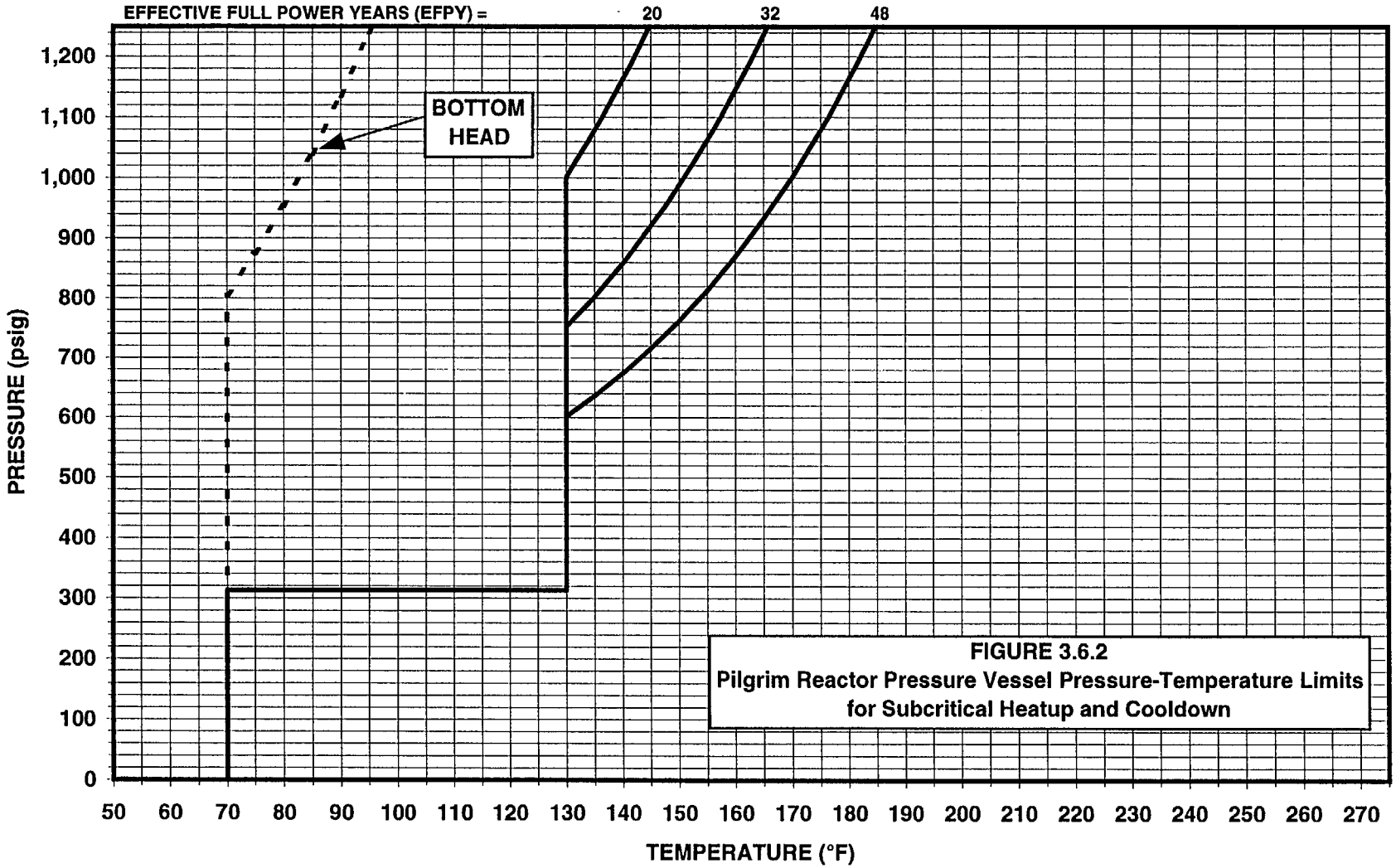

FIGURE 3.6.2
Pilgrim Reactor Pressure Vessel Pressure-Temperature Limits for Subcritical Heatup and Cooldown

PILGRIM REACTOR VESSEL PRESSURE-TEMPERATURE LIMITS
CRITICAL CORE OPERATION

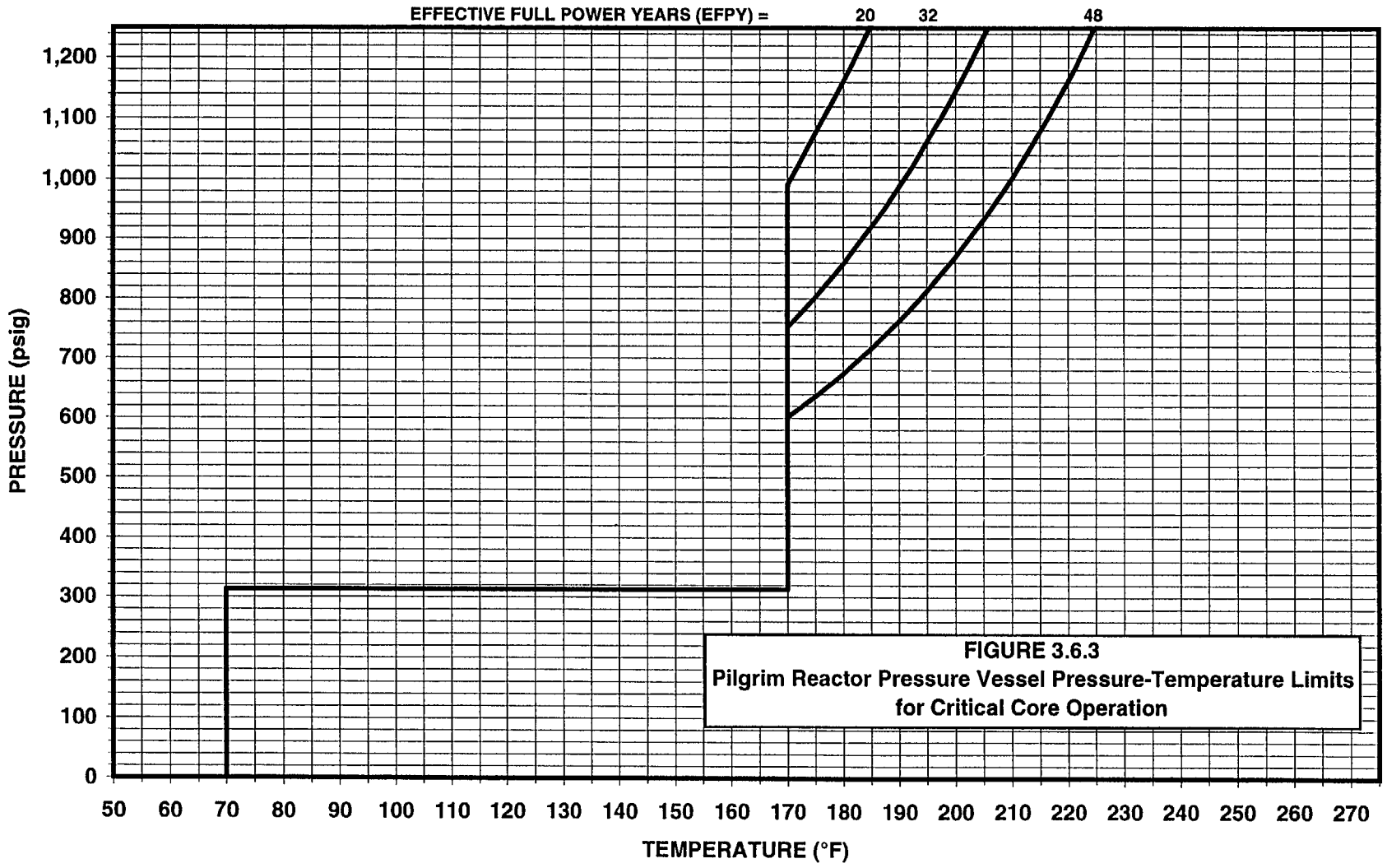

FIGURE 3.6.3
Pilgrim Reactor Pressure Vessel Pressure-Temperature Limits for Critical Core Operation

BASES:

3/4.6 PRIMARY SYSTEM BOUNDARY (Cont)

A. Thermal and Pressurization Limitations (Cont)

The bottom head, defined as the spherical portion of the reactor vessel located below the lower circumferential weld, was also evaluated. Reference transition temperatures (RT_{ndt}) were developed for the bottom head and the resulting pressure vs. temperature curves plotted on Figures 3.6-1 and 3.6-2. It has been determined that the bottom head temperatures are allowed to lag the vessel shell temperatures (Reference: Structural Integrity Associates (SIA) Report SIR-00-108, dated September 11, 2000). The referenced analysis utilizes the stress results established in the Combustion Engineering Inc., Pilgrim Reactor Vessel Design Report, No. CENC 1139, dated 1971, and combines the stress analysis results, specific to the bottom head, with the pressurization temperatures necessary to maintain fracture toughness requirements in accordance with the ASME Boiler and Pressure Vessel Code, Section III, the criteria of 10CFR50, Appendix G, and the supplementary guidelines of Reg. Guide 1.99, Rev. 2.

For Pilgrim pressure-temperature restrictions, two locations in the reactor vessel are limiting. The closure region controls at lower pressures and the beltline controls at higher pressures.

The nil-ductility transition (NDT) temperature is defined as the temperature below which ferritic steel breaks in a brittle rather than ductile manner. Radiation exposure from fast neutrons (>1 Mev) above about 10^{17} nvt may shift the NDT temperature of the vessel metal above the initial value. Impact tests from the first material surveillance capsule removed at 4.17 EFPY indicated a maximum RT_{ndt} shift of 55 degrees F for the weld specimens.

Neutron flux wires and samples of vessel material are installed in the reactor vessel adjacent to the vessel wall at the core midplane level. The wires and samples will be periodically removed and tested to experimentally verify the values used for Figures 3.6-1, 3.6-2, and 3.6-3. The withdrawal schedule of Table 4.6-3 has been established as required by 10CFR50, Appendix H.

The RT_{ndt} of the closure region is +10 degrees F. The initial RT_{ndt} for the beltline weld and base metal are -48 degrees F and 0 degrees F, respectively. These RT_{ndt} temperatures are based upon unirradiated test data, adjusted for specimen orientation in accordance with USNRC Branch Technical Position MTEB 5-2.

The closure and bottom head regions are not exposed to neutron fluence (> 1 Mev) over the vessel life sufficient to cause a shift in RT_{ndt}. The pressure-temperature limitations (Figures 3.6-1, 3.6-2, and 3.6-3) of the closure and bottom head regions will therefore remain constant throughout vessel life. Only the beltline region of the reactor vessel will experience a shift in RT_{ndt} with a resultant increase in pressure-temperature limits.

The curves apply to 100% bolt preload condition but are conservative for lesser bolt preload conditions.

BASES:

3/4.6 PRIMARY SYSTEM BOUNDARY (Cont)

A. Thermal and Pressurization Limitations (Cont)

For critical core operation when the water level is within the normal range for power operation and the pressure is less than 20% of the preservice system hydrostatic test pressure (313 psi), the minimum permissible temperature of the highly stressed regions of the closure flange is RT_{ndt} + 60 degrees F = 70 degrees F; thus, a cutoff limit of 70 degrees F was chosen as shown on Figure 3.6-3 and as required by 10CFR50 Appendix G, paragraph IV. A.3. This same cutoff is conservatively included in the limits for hydrostatic and leak tests and for non-critical operation, as shown on Figures 3.6-1 and 3.6-2, respectively, in order to be consistent with the limits for critical operation.

The closure region is more limiting than the feedwater nozzle with respect to both stress intensity and RT_{ndt}. Therefore, the pressure-temperature limits of the closure are controlling.

The minimum bolt-up temperature is the minimum allowable nil ductility transition temperature (RT_{ndt}) at pressures below the 20% of pre-operational system hydrostatic test pressure that bolt pre-load stress can be applied to the reactor vessel closure region. It is defined as the initial RT_{ndt} of the higher stressed component of the reactor vessel that includes the vessel head, head flange and shell adjacent to the vessel flange. The maximum RT_{ndt} is + 10 degrees F. For conservatism a minimum bolt-up temperature of 55 degrees F is chosen because this temperature provides sufficient margin between the lowest service temperature at 20% of the pre-operational hydrostatic test pressure prior to pressurization.

The adjusted reference temperature shift is based on Regulatory Guide 1.99, Revision 2, dated May 1988; the analytical results of General Electric Report MDE 277-1285, Revision 1, dated January 21, 1985, regarding projected neutron fluence; and Structural Integrity Associates (SIA) report SIR-00-108 dated September 11, 2000, for RT_{ndt} versus fluence as a function of temperature and pressure.

B. Coolant Chemistry

The reactor vessel coolant chemistry requirements are discussed in Subsection 4.2 of the FSAR.

A radioactivity concentration of 20 μ Ci/ml total iodine can be reached if there is significant fuel failure or if there is a failure or a prolonged shutdown of the cleanup demineralizer. Calculations performed by the AEC staff for this activity level results in a radiological dose at the site boundary of 8 rem to the thyroid from a postulated rupture of a main steam line assuming a 5 second valve closing time and a coolant inventory release of 3×10^4 lbs.

A reactor sample will be used to assure that the limit of Specification 3.6.B.1 is not exceeded.

ATTACHMENT 3 TO PILGRIM LETTER

Marked-up Current Pilgrim
Technical Specification Figures 3.6.1, 3.6.2, 3.6.3 and Bases

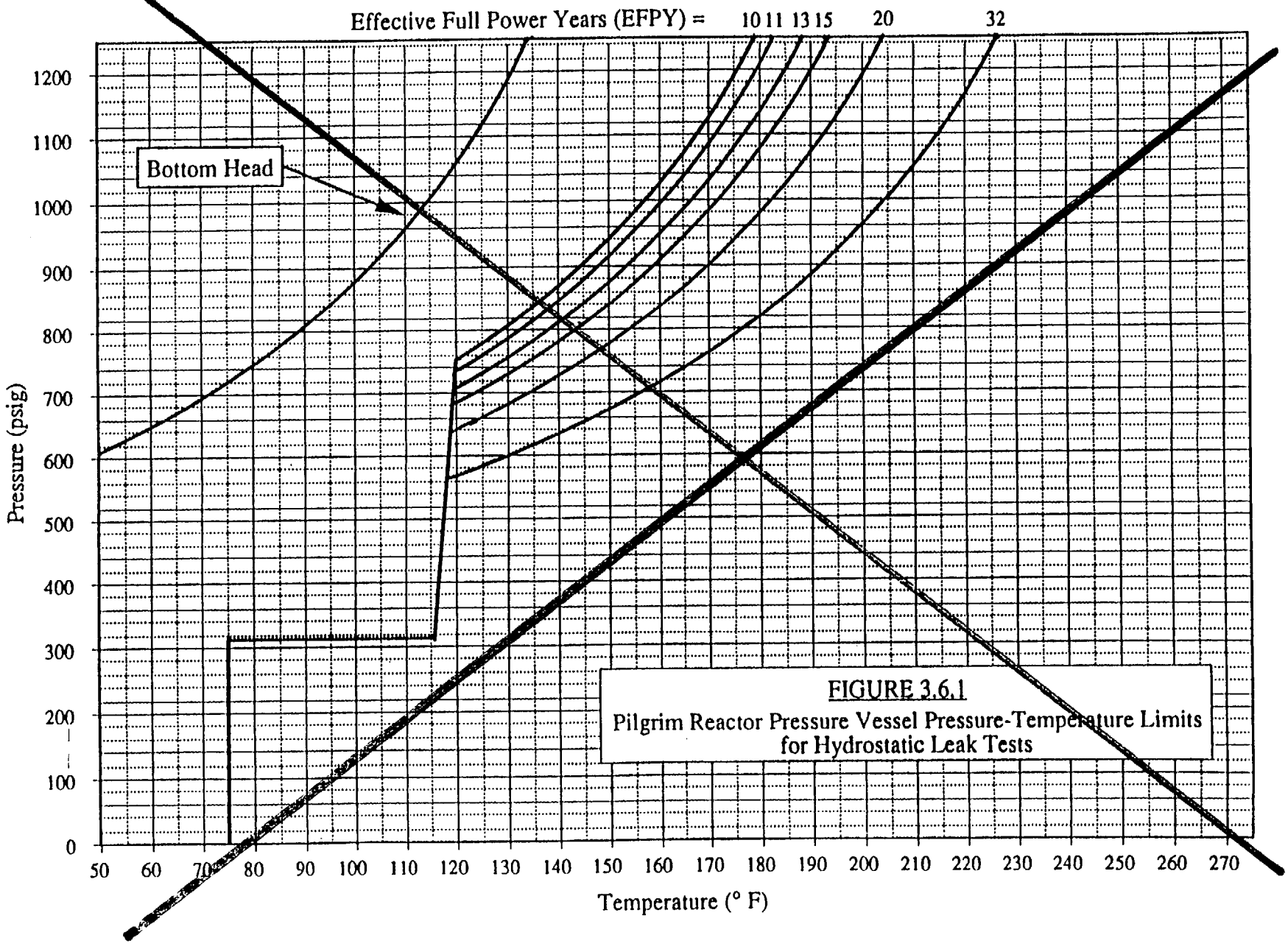

FIGURE 3.6.1
Pilgrim Reactor Pressure Vessel Pressure-Temperature Limits for Hydrostatic Leak Tests

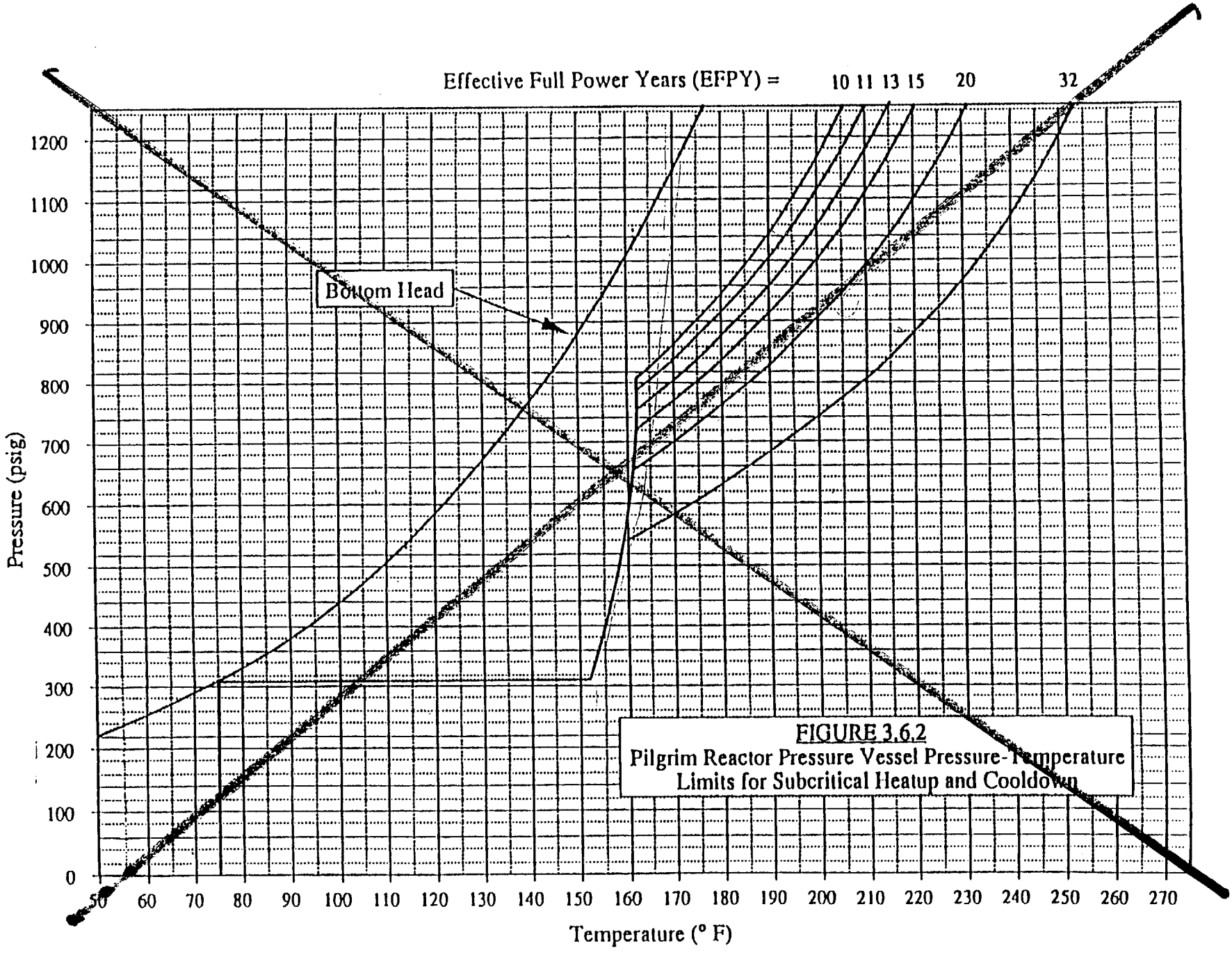

FIGURE 3.6.2
Pilgrim Reactor Pressure Vessel Pressure-Temperature Limits for Subcritical Heatup and Cooldown

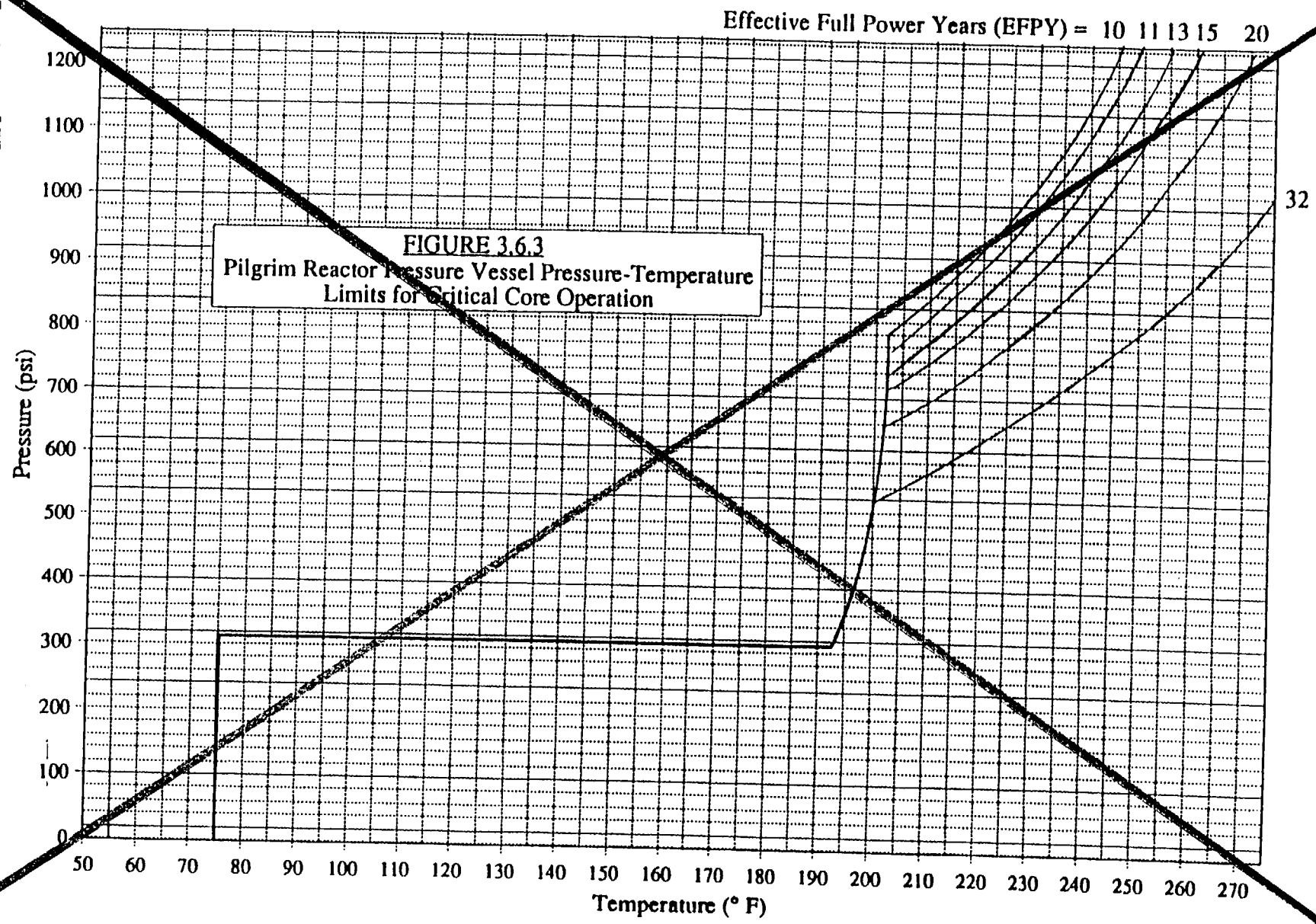

FIGURE 3.6.3
Pilgrim Reactor Pressure Vessel Pressure-Temperature Limits for Critical Core Operation

BASES:

3/4.6 PRIMARY SYSTEM BOUNDARY (Cont)

A. Thermal and Pressurization Limitations (Cont)

The bottom head, defined as the spherical portion of the reactor vessel located below the lower circumferential weld, was also evaluated. Reference transition temperatures (RT_{NDT}) were developed for the bottom head and the resulting pressure vs. temperature curves plotted on Figures 3.6-1 and 3.6-2. It has been determined that the bottom head temperatures are allowed to lag the vessel shell temperatures, Reference: ~~Teledyne Engineering Services (TES) report TR-6051C-1, dated June 27, 1986~~ Structural Integrity Associates (SIA) Report SIR-00-108, dated September 11, 2000. The referenced analysis utilizes the stress results established in the Combustion Engineering Inc., Pilgrim Reactor Vessel Design Report, No. CENC 1139, dated 1971, and combines the stress analysis results, specific to the bottom head, with the pressurization temperatures necessary to maintain fracture toughness requirements in accordance with the ASME Boiler and Pressure Vessel Code, Section III, the criteria of 10CFR50, Appendix G, and the supplementary guidelines of Reg. Guide 1.99, Rev. 2.

For Pilgrim pressure-temperature restrictions, two locations in the reactor vessel are limiting. The closure region controls at lower pressures and the beltline controls at higher pressures.

The nil-ductility transition (NDT) temperature is defined as the temperature below which ferritic steel breaks in a brittle rather than ductile manner. Radiation exposure from fast neutrons (>1 mev) above about 10^{17} nvt may shift the NDT temperature of the vessel metal above the initial value. Impact tests from the first material surveillance capsule removed at 4.17 EFPY indicated a maximum RT_{NDT} shift of 55°F for the weld specimens.

Neutron flux wires and samples of vessel material are installed in the reactor vessel adjacent to the vessel wall at the core midplane level. The wires and samples will be periodically removed and tested to experimentally verify the values used for Figures 3.6-1, 3.6-2, and 3.6-3. The withdrawal schedule of Table 4.6-3 has been established as required by 10CFR50, Appendix H.

The RT_{NDT} of the closure region is **+10** ~~-5~~°F. The initial RT_{NDT} for the beltline weld and basemetal are **-48** ~~-50~~°F and 0°F respectively. These RT_{NDT} temperatures are based upon unirradiated test data, adjusted for specimen orientation in accordance with USNRC Branch Technical Position MTEB 5-2.

The closure and bottom head regions are not exposed to neutron fluence (> 1 Mev) over the vessel life sufficient to cause a shift in RT_{NDT}. The pressure-temperature limitations (Figures 3.6-1, 3.6-2, and 3.6-3) of the closure and bottom head regions will therefore remain constant throughout vessel life. Only the beltline region of the reactor vessel will experience a shift in RT_{NDT} with a resultant increase in Pressure-Temperature limits.

The curves apply to 100% bolt preload condition, but are conservative for lesser bolt preload conditions.

A. Thermal and Pressurization Limitations (Cont)

For critical core operation when the water level is within the normal range for power operation and the pressure is less than 20% of the preservice system hydrostatic test pressure (313 psi), the minimum permissible temperature of the highly stressed regions of the closure flange is RT_{NDT} + 60 degrees F = 70 degrees F; thus a cutoff limit of 70 degrees F was chosen as shown on Figure 3.6-3 and as permitted by 10CFR50 Appendix G, paragraph IV. A.3. This same cutoff is conservatively included in the limits for hydrostatic and leak tests and for non-critical operation, as shown on Figures 3.6-1 and 3.6-2 respectively, in order to be consistent with the limits for critical operation.

The closure region is more limiting than the feedwater nozzle with regards to both stress intensity and RT_{NDT}. Therefore the pressure-temperature limits of the closure are controlling.

The minimum bolt-up temperature is the minimum allowable nil ductility transition temperature (RT_{ndt}) at pressures below the 20% of pre-operational system hydrostatic test pressure that bolt pre-load stress can be applied to the reactor vessel closure region. It is defined as the initial RT_{ndt} of the higher stressed component of the reactor vessel that includes the vessel head, head flange and shell adjacent to the vessel flange. The maximum RT_{ndt} is + 10 degrees F. For conservatism a minimum bolt-up temperature of 55 degrees F is chosen because this temperature provides sufficient margin between the lowest service temperature at 20% of the pre-operational hydrostatic test pressure prior to pressurization.

The adjusted reference temperature shift is based on Regulatory Guide 1.99, Revision 2, dated May 1988; the analytical results of General Electric Report MDE 277-1285, Revision 1, dated January 21, 1985, regarding projected neutron fluence; and Structural Integrity Associates (SIA) report SIR-00-108 dated September 11, 2000 ~~Teledyne Engineering Services Reports, TR-6052B-1, Revision 1, dated June 26, 1986, as supplemented by TR-7487, dated April 16, 1991,~~ for RT_{NDT} vs. fluence as a function of temperature and pressure~~, and TR-6052C-1, dated June 27, 1986, for the RPV bottom head pressurization temperatures.~~

B. Coolant Chemistry

The reactor vessel coolant chemistry requirements are discussed in Subsection 4.2 of the FSAR.

A radioactivity concentration of 20 μ Ci/ml total iodine can be reached if there is significant fuel failure or if there is a failure or a prolonged shutdown of the cleanup demineralizer. Calculations performed by the AEC staff for this activity level results in a radiological dose at the site boundary of 8 rem to the thyroid from a postulated rupture of a main steam line assuming a 5 second valve closing time and a coolant inventory release of 3×10^4 lbs.

A reactor sample will be used to assure that the limit of Specification 3.6.B.1 is not exceeded.