U.S. NUCLEAR REGULATORY COMMISSION

STANDARD REVIEW PLAN

OFFICE OF NUCLEAR REACTOR REGULATION

5.3.2 PRESSURE-TEMPERATURE LIMITS AND PRESSURIZED THERMAL SHOCK[1]

REVIEW RESPONSIBILITIES

Primary - ~~Materials Engineering Branch (MTEB)~~ Materials and Chemical Engineering Branch (EMCB)[2]

Secondary - None

I. AREAS OF REVIEW

1. Pressure-Temperature Limits

The regulations requiring the imposition of pressure-temperature limits on the reactor coolant pressure boundary are the following:

> ~~Paragraph~~Section[3] 50.55a of 10 CFR Part 50, "Codes and Standards," requires that structures, systems, and components (SSC)[4] be designed, fabricated, erected, constructed, tested, and inspected to quality standards commensurate with the importance of the safety function to be performed. In addition, General Design Criterion 1 of Appendix A of 10 CFR Part 50, "Quality Standards and Records," requires that the codes and standards used to assure quality products in keeping with the safety function be identified and evaluated to determine their adequacy.
>
> General Design Criterion 14 of Appendix A of 10 CFR Part 50, "Reactor Coolant Pressure Boundary," requires that the reactor coolant pressure boundary be designed, fabricated, erected, and tested in order to have an extremely low probability of abnormal leakage, of rapid failure, and of gross rupture. Likewise, General Design Criterion 31, "Fracture Prevention of Reactor Coolant Pressure Boundary," requires, in part, that the

reactor coolant pressure boundary be designed with sufficient margin to assure that when stressed under operating, maintenance and testing, the boundary behaves in a non-brittle[5] manner and the probability of rapidly propagating fracture is minimized. Further, in order to assess the structural integrity of the reactor vessel, General Design Criterion 32, "Inspection of Reactor Coolant Pressure Boundary," requires, in part, an appropriate materials surveillance program for the reactor vessel beltline region.

The reactor vessel beltline materials for Pressurized Water Reactors (PWRs) are evaluated for susceptibility to pressurized thermal shock (PTS) through review of 1) the reference temperature, RT_{PTS}, calculations and screening criterion, and, if the RT_{PTS} value is projected to exceed the PTS screening criterion before the expiration date of the license, 2) any associated safety analyses performed to support reactor operation.[6]

The pressure-temperature limits imposed on the reactor coolant pressure boundary during any condition of normal operation, including anticipated operational occurrences, and hydrostatic[7] tests are reviewed in this section of the Standard Review Plan (SRP) to assure adequate safety margins of structural integrity for the ferritic components of the reactor coolant pressure boundary.

Review Interfaces:

EMCB also reviews the material characteristics of the reactor coolant pressure boundary and reactor vessel, including fracture toughness properties, as part of its review responsibilities for SRP Sections 5.2.3 and 5.3.1.

In addition, the EMCB will coordinate with the Reactor Systems Branch (SRXB) which reviews the over-pressure protection system for consistency with the Appendix G pressure-temperature limits as part of its review responsibility for SRP Section 5.2.2 and Branch Technical Position (BTP), BTP RSB 5-2.[8] The SRXB also reviews the peak reactor vessel wall fluence for the design life of the plant as part of its review responsibility for SRP Section 4.3.[9]

II. ACCEPTANCE CRITERIA[10]

The acceptability of the reactor coolant pressure boundary pressure-temperature limits is based on meeting the relevant requirements of the following Commission regulations:

A. 10 CFR Part 50, Appendix A, General Design Criterion 1, "Quality Standards and Records," as it relates to quality standards for design, fabrication, erection and testing;

B. 10 CFR Part 50, Appendix A, General Design Criterion 14, "Reactor Coolant Pressure Boundary," as it relates to assuring an extremely low probability of abnormal leakage, rapidly propagating failure and gross rupture of the reactor coolant pressure boundary;

C. 10 CFR Part 50, Appendix A, General Design Criterion 31, "Fracture Prevention of Reactor Coolant Pressure Boundary," as it relates to assuring that the reactor coolant pressure boundary will behave in a non-brittle manner and the probability of rapidly propagating fracture is minimized;

D. 10 CFR Part 50, Appendix A, General Design Criterion 32, "Inspection of Reactor Coolant Pressure Boundary," as it relates to the reactor vessel materials surveillance program;

E. 10 CFR Part 50, §50.55a, "Codes and Standards", as it relates to quality standards for design, and determination and monitoring of material fracture toughness;

F. 10 CFR Part 50, §50.60, "Acceptance Criteria for Fracture Prevention Measures for Lightwater Nuclear Power Reactors for Normal Operation," as it relates to compliance with the requirements of 10 CFR 50, Appendices G and H;[11]

G. 10 CFR Part 50, §50.61, "Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events," as it relates to fracture toughness criteria for PWRs relevant to pressurized thermal shock events; and[12]

H. 10 CFR Part 50, Appendix G, "Fracture Toughness Requirements," as it relates to material testing and fracture toughness.

~~The requirements of paragraph 50.55a and General Design Criteria 1, 14, 31 and 32 of Appendix A of 10 CFR Part 50 are met by the assurance that material of the reactor coolant pressure boundary possess adequate fracture toughness properties to resist rapidly propagating failure and act in a nonbrittle manner when stressed under operating, maintenance, testing, and anticipated operational conditions. The requirement, in part, of General Design Criterion 32 is met by conducting a surveillance program to monitor the change in fracture toughness properties of the ferritic materials in the reactor vessel.~~

~~The fracture toughness requirements for ferritic materials in the pressure- retaining components of the RCPB are specified for testing and operational conditions, including anticipated operational occurrences, in Section IV of Appendix G of 10 CFR Part 50. This appendix requires the acceptance and performance criteria of Appendix G of Section III of the ASME Boiler and Pressure Vessel Code. Pressure-temperature calculation procedures are described in Appendix G of the ASME code; while the detailed technical basis for the ASME code requirement is provided by the Welding Research Council (WRC) Bulletin 175, "PVRC Recommendation on Toughness Requirements for Ferritic Materials." Changes in the fracture-toughness properties of materials in the beltline region, resulting from neutron irradiation and the thermal environment, are monitored by a surveillance program in compliance to the requirements of Appendix H of 10 CFR Part 50. The effect of neutron fluence on the shift in the nil-ductility temperature of pressure vessel steel is predicted by Regulatory Guide 1.99, "Effect of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials."~~

Specific criteria necessary to meet the relevant requirements of the Commission's regulations listed above are as follows:[13]

1. <u>Applicable Regulations, Codes, and Basis Documents</u>

   10 CFR 50.60, and associated [14]Appendices G and H of 10 CFR Part 50,[15] describe the conditions that require pressure-temperature limits and provide the general basis for these

limits. These appendices specifically require that pressure-temperature limits must provide safety margins at least as great as those recommended in the ASME Boiler and Pressure Vessel Code (hereinafter "the Code"), Section III, Appendix G, "Protection Against Nonductile Failure," (Reference 14)[16] during heatup, cooldown, and test conditions. Appendix G to 10 CFR Part 50 also requires additional safety margins whenever the reactor core is critical (except for low-level physics tests).

In addition to the pressure-temperature limits established in accordance with 10 CFR 50.60, and Appendix G, projected values of RT_{PTS} must be determined for PWR reactor vessel beltline materials in accordance with 10 CFR 50.61. These values of RT_{PTS} must meet the temperature limit screening criteria in the rule, or must be accompanied by safety analyses developed in accordance with 10 CFR 50.61 and the guidance of Regulatory Guide 1.154.[17]

2. Technical Bases

Since many of the fracture toughness requirements for the ferritic materials in the pressure-retaining components were not required at the time some of the reactor facilities were designed and constructed, the ~~Materials Engineering Branch~~Materials and Chemical Engineering Branch Technical Position ~~MTEB~~ EMCB[18] 5-2, "Fracture Toughness Requirements," describe procedures for making estimates and assumptions on the fracture toughness properties of materials in the older plants. Calculations are required, and an evaluation is made by the reviewer to show compliance with the regulations and to show an adequate margin of quality and safety for the facility. When it has been determined that certain requirements of Appendices G or H have not been strictly complied with by these older plants, and when it has been determined that an equivalent level of quality and safety, as required by the regulations exist, then exemption to the specific requirements of these appendices will be granted by the Commission as described in 10 CFR 50.60 and 10 CFR 50, Appendix G, paragraph III.A[19].

a. The principles of linear elastic fracture mechanics (LEFM) are used to determine safe operational conditions. The basic parameter of LEFM is the stress intensity factor, K_I, which is a function of the stress state and flaw configuration. An analytical method is used to determine the effects of real or postulated flaws. The minimum K_I that can cause failure is defined as the critical stress intensity factor, K_{Ic}, and is the material parameter used in this method. The K_{Ic} of the material is either directly measured as a function of temperature, or is conservatively estimated, using information from other fracture toughness tests.

b. The Code specifies the maximum K_{Ic}, as a function of temperature, that can be assumed for the specific material, based on results of tests on the material used. This value is called $K_{I\cancel{R}a}$,[20] reference stress intensity factor. The Code also provides rules for calculating K_I, including definitions of postulated flaws, and specifies the safety factors to be applied. The acceptance criterion is that the $K_{I\cancel{R}a}$[21] of the material must always be higher than the K_I calculated.

c. Direct measurement of the K_{Ic} as a function of temperature is expensive and time consuming and requires more sample material than is usually available. Correlations between the K_{Ic} determined directly and results of simpler fracture toughness tests are not exact, but may be used if appropriate allowances are made for variations in material behavior and data scatter. The Code gives values of $K_{I\cancel{R}a}$[22] as a function of temperature relative to a conservative determination of the reference temperature of the material. This reference temperature, RT_{NDT}, is determined for the ferritic materials of components for which operating and testing limit curves must be calculated. The effects of radiation on the fracture toughness of the material in the beltline region of the reactor vessel is accounted for by adjusting the RT_{NDT} of the affected material upward. The amount of upward shift depends on the composition of the steel (especially its copper and ~~phosphorous~~nickel[23] content), and the neutron fluence. Conservative predictions of the effect of radiation on the RT_{NDT} based on ~~data~~the methods[24] in Regulatory Guide 1.99, are factored into the original limit curves. The continued conservatism of these predictions throughout plant life is verified by a mandatory material surveillance program described in Appendix H of 10 CFR Part 50.

d. The Code specifies the stress components that must be used for the K_I calculations, and the factors that must be applied to each to provide adequate safety margins. The Code, by reference to WRC Bulletin~~-~~[25] 175 (Reference 15)[26], specifies the expression to use for calculating the K_I, using the applied stresses and the postulated flaw geometry. Although calculations are usually made by a computer, curves are provided in the Code to facilitate the use of conservative hand calculations if desired.

3. <u>Pressure-Temperature Requirements</u>

The requirements for the pressure-temperature limits are as follows:

a. Pressure-Temperature Limits for Preservice Hydrostatic Tests

During preservice hydrostatic tests (if fuel is not in the vessel), the $K_{I\cancel{R}a}$[27] must be greater than the K_I caused by pressure. The expression used is:

$$K_I = K_I(\text{pressure}) < K_{I\cancel{R}a}$$ [28]

b. Pressure-Temperature Limits for Inservice Leak and Hydrostatic Tests

During performance of inservice leak and hydrostatic tests, the $K_{I\cancel{R}a}$[29] must be greater than 1.5 times the K_I caused by pressure. The expression used is:

$$K_I = 1.5\ K_I(\text{pressure}) < K_{I\cancel{R}a}$$ [30]

c. Pressure-Temperature Limits for Heatup and Cooldown Operations

At all times during heatup and cooldown operations, the $K_{I\cancel{R}a}$[31] must be greater than the sum of 2 times the K_I caused by pressure and the K_I caused by thermal gradients. The expression used is:

$$K_I = 2K_I(\text{pressure}) + K_I(\text{thermal}) < K_{I\cancel{R}a}$$ [32]

d. Pressure-Temperature Limits for Core Operation

At all times that the reactor core is critical (except for low power physics tests) the temperature must be higher than that required for inservice hydrostatic testing, and in addition, the pressure-temperature relationship shall provide at least a 22°C (40°F)[33] margin over that required for heatup and cooldown operations.

4. Material Reference Temperature Limits for PWR Pressurized Thermal Shock Events

Values of RT_{PTS} projected using the methods of 10 CFR 50.61 for the time of the initial application submittal and for the projected expiration date of the operating license must not exceed the screening criteria of 132°C (270°F) for plates, forgings, and axial weld materials, and 149°C (300°F) for circumferential weld materials. For RT_{PTS} values projected to exceed the screening criteria, safety analyses must be provided that include proposed flux reduction programs or other corrective actions to prevent potential PTS related failure of the reactor vessel if continued plant operation beyond the screening criteria is planned.[34]

Technical Rationale:[35]

The technical rationale for application of the above acceptance criteria to the reactor coolant pressure boundary pressure-temperature limits is discussed in the following paragraphs.

1. GDC 1 and 10 CFR 50.55a establish quality assurance requirements for the design, fabrication, erection, and testing of structures, systems and components (SSC) important to safety. GDC 1 establishes that the quality standards to be applied to SSC shall be commensurate with the importance of the safety functions to be performed. 10 CFR 50.55a, in relevant part, establishes those provisions of the ASME Boiler and Pressure Vessel Code that must be complied with to ensure that SSCs are designed, fabricated, erected, tested, and inspected to quality standards commensurate with the importance of the safety function to be performed. The primary safety functions of the RCPB are to prevent a loss of reactor coolant through leakage or gross failure of RCPB piping or components, and to act as a containment barrier to the release of fission products in the event of an accident resulting in fuel damage. Pressure-temperature limits are established for the reactor coolant pressure boundary (RCPB) in accordance with Section III of the ASME Boiler and Pressure Vessel Code, Appendix G, to ensure that the RCPB material fracture toughness requirements are satisfied. Compliance with GDC 1 and 10 CFR 50.55a provides assurance that the RCPB meets the appropriate quality standards of the

ASME Code and thus the probability of RCPB material failure and the subsequent effects on reactor core cooling and confinement are minimized.

2. GDC 14 establishes that the RCPB be designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture. The RCPB provides for confinement of reactor coolant and acts as a barrier to the release of fission products in the event of an accident resulting in fuel failure. Pressure-temperature limits established for the RCPB ensure that the material fracture toughness requirements for the RCPB piping and components are met and that the RCPB will act in a non-brittle manner under operating, maintenance, testing, and postulated accident conditions. Application of GDC 14 to the RCPB, with regard to the pressure-temperature limits, provides assurance that the RCPB meets the material fracture toughness requirements and will act in a non-brittle manner, thereby providing a low probability of significant degradation or gross failure of the RCPB that could cause a loss of reactor coolant inventory, and a reduction in the capability to confine fission products.

3. GDC 31 establishes that the RCPB be designed with sufficient margin to assure that when stressed under operating, maintenance, testing, and postulated accident conditions, the boundary behaves in a non-brittle manner and the probability of rapidly propagating fracture is minimized. The design is required to reflect consideration of service temperatures and other conditions of the boundary material and the uncertainties in determining material properties, the effects of irradiation on material properties, residual, steady state and transient stresses, and size of flaws. The RCPB provides a fission product barrier, confinement of reactor coolant, and flow paths to facilitate core cooling. Regulatory Guide 1.99 provides methods for predicting irradiation effects on fracture toughness properties that are applicable to compliance with the requirements of GDC 31. Application of GDC 31 assures that the pressure-temperature limits for the RCPB are appropriately determined and provide sufficient margin to account for uncertainties associated with flaws and the effects of service and operating conditions, and thereby provide a minimum probability of brittle material behavior leading to rapidly propagating failure. The probability of substantial reduction in capability to contain reactor coolant inventory, reduction in capability to confine fission products, and interference with core cooling is thereby minimized.

4. GDC 32 requires a material surveillance program for the reactor pressure vessel. Changes in the fracture-toughness properties of materials in the reactor pressure vessel beltline region, resulting from neutron irradiation and the thermal environment, are monitored by a surveillance program in compliance with the requirements of Appendix H of 10 CFR Part 50. Surveillance program compliance with GDC 32 and Appendix H is reviewed under SRP Section 5.3.1. Compliance with GDC 32 provides assurance that pressure-temperature limits continue to provide sufficient margin to minimize the probability of rapidly propagating failure of the RPV throughout the plant lifetime (see Technical Rationale for GDC 31).

5. 10 CFR 50.60 requires that all light-water nuclear power reactors must meet the fracture toughness requirements, including pressure-temperature limits, as set forth in 10 CFR 50,

Appendix G. Compliance with the requirements of this rule and Appendix G provides assurance regarding the structural integrity of the RCPB and specifically the reactor vessel. The Technical Rationale for this rule is established under the Technical Rationale discussion for 10 CFR 50, Appendix G, below.

6. 10 CFR 50.61, establishes fracture toughness requirements for protection against pressurized thermal shock (PTS) events. Pressurized thermal shock events involve transients in pressurized water reactors that cause severe overcooling in conjunction with overpressurization. The thermal stresses in combination with the pressure stresses increase the potential for brittle fracture in the presence of an initiating flaw in low toughness material. This material may be present in the reactor vessel beltline where neutron radiation gradually embrittles the material over the plant lifetime. The PTS rule provides calculational methods and acceptance criteria for determining the effect of embrittlement on the reactor vessel materials and establishing the material reference temperature limits beyond which continued operation of the plant must be justified by corrective actions and plant-specific safety analyses. Establishing, monitoring and maintaining the structural integrity of the reactor vessel materials is essential in protecting against a failure of the RCPB and the subsequent loss of core cooling and fission product containment. Compliance with the requirements of 10 CFR 50.61 provides assurance that the reactor vessel materials will not be subject to failure from PTS during the life of the reactor.

7. 10 CFR 50, Appendix G, establishes that the ferritic pressure-retaining components of the RCPB meet requirements for fracture toughness during system hydrostatic tests and any condition of normal operation, including anticipated operational occurrences. Fracture toughness properties of ferritic materials increase significantly above the point referred to as the nil-ductility transition temperature. This temperature is established for the RCPB material in accordance with Section III of the ASME Code as supplemented by the requirements of 10 CFR 50, Appendix G. Pressure-temperature limits established in accordance with the ASME Code and Appendix G, are used to establish operating parameters that provide assurance that the RCPB will act in a non-brittle manner when subjected to stresses associated with normal operations, maintenance, testing and anticipated operational occurrences. The pressure-temperature limits must be monitored and adjusted to account for the effects of radiation embrittlement of the RCPB materials over the life of the plant, particularly the materials in the reactor vessel beltline. This is facilitated by a material surveillance program consistent with 10 CFR 50, Appendix H. Compliance with the requirements of Appendix G provides a method of satisfying the requirements of GDCs 14 and 31 with regard to assuring that the RCPB acts in a non-brittle manner and that the probability of rapidly propagating failure and gross rupture of the RCPB is extremely low.

III. REVIEW PROCEDURES

The reviewer will select and emphasize material from the procedures described below, as may be appropriate for a particular case.

1. ~~Preliminary Safety Analysis Report (PSAR)~~Construction Permit/Design Certification Reviews[36]

Information in the Preliminary Safety Analysis Report (PSAR)/Standard Safety Analysis Report (SSAR)[37] is reviewed for a commitment that the fracture toughness of the ferritic materials in the reactor coolant pressure boundary will comply with the requirements of Appendix G of 10 CFR Part 50, as detailed in Section III of the ASME Boiler and Pressure Vessel Code and that the materials in the beltline region of the reactor vessel will comply with the requirements of 10 CFR 50.60 and associated[38] Appendices G and H of 10 CFR Part 50, 10 CFR 50.61 (PWRs only),[39] and the guidance of[40] Regulatory Guide 1.99~~, "Effect of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials.~~~~"~~[41]

2. ~~Final Safety Analysis Report (FSAR)~~Operating License/Combined License Reviews[42]

The limits in the plant Technical Specifications will be shown using real temperature. These curves and their bases are reviewed to determine acceptability in the following areas:

a. The limiting RT_{NDT} has been properly determined, and radiation effects are included in a conservative manner.

b. Limits are shown for all required conditions.

c. The limits proposed are consistent with the acceptance criteria described in II. above.

d. The procedures for updating the limit curves, in conjunction with scheduled tests on material surveillance specimens, are well defined and included in the Technical Specifications.

3. Acceptability Determination Methods

The reviewer evaluates each limit curve for acceptability by performing check calculations using the simplified methods referenced in the Code and WRC Bulletin 175 that have been verified by the ~~Materials Engineering Branch~~ Materials and Chemical Engineering Branch[43] to yield conservative values. These methods are described in detail by examples below, and the curves necessary to perform the calculations are included herein as Figures 1, 2 and 3.[44]

a. Preservice Hydrostatic Tests

The preservice hydrotest at 1.25 times the[45] design pressure corresponds to the standard Code component hydrotest usually performed in the shop, but in this case it is the hydrotest for field welds, so it may involve the entire reactor coolant system.

The Code recommends that component hydrostatic tests be run at a temperature no lower than $RT_{NDT} + 33°C$ ($RT_{NDT} + 60°F$)[46], but also recommends that system tests should have more stringent requirements. The ~~MTEB~~EMCB[47] position is that the minimum temperature for the preservice test, if fuel is not in the vessel, be determined using the methods of Code Section III, Appendix G, using less stringent factors.

First, the RT_{NDT} of the vessel material must be determined. This is defined by the Code for new plants, and is essentially a conservative value of the nil-ductility transition temperature (NDTT)[48] as determined by drop weight test. Guidelines for estimating the RT_{NDT} if the prescribed tests have not been run are given by Branch Technical Position - ~~MTEB~~ EMCB[49] 5-2, "Fracture Toughness Requirements." Technical justification for all estimates of RT_{NDT} must be provided by the applicant.

The toughness of the material is a function of the difference between the RT_{NDT} of the material and the temperature of interest. The Code provides a curve (Figure G-2210.1) for the allowable calculated stress intensity factor ($K_{I\cancel{R}a}$[50]) as a function of the temperature relative to RT_{NDT}. Refer to Figure 2 herein.

The Code also provides a recommended basis for calculating K_I, including recommendations for assumed flaw size and shape, and appropriate front and back surface correction factors. Because the assumed flaw size is proportional to the wall thickness, t (flaw depth = 0.25 t and length = 1.5 t), the K_I expressions are simplified to multiples that are a function only of wall thickness and stress level. These factors, M_m for membrane stresses and M_B for bending stresses, are provided in graphical form in Figure G-2214.1 of the Code. Refer to Figure 1 herein.

The criterion recommended by ~~MTEB~~ EMCB[51] can be expressed as

$K_I < K_{I\cancel{R}a}$[52] for the shell region.

To get K_I, the stress level and wall thickness must be known. The pressure for the hydrostatic test is 1.25 times the design pressure, so that the higher of two simple methods described below to approximate the membrane stress should be accurate enough for this purpose:

stress = 1.25 times the Code allowable (S_m)

$$\text{stress} = \frac{Pr}{t}$$

where P is the test pressure and r is the vessel radius. As an example, assume a vessel with a design pressure of 2500 psig, made of steel with an S_m of 26,700

psi, and a minimum yield strength of 50,000 psi. The stress for the preservice hydrotest is then

$$26,700 \times 1.25 = 33,400 \text{ psi, or}$$

$\frac{(1.25)(2500)(95)}{9} = 33,\cancel{4}000$[53] psi, for a vessel with a radius of 95 inches and a wall thickness of 9 inches.

The next step is to determine the factor to apply to this stress to obtain K_I. Figure G-2214.1 (reproduced here as Fig. 1) provides several curves, depending on the ratio of the stress level to the yield strength of the material. In this case, the conservative (i.e., higher of the above two approximations)[54] stress level is 33,400; the yield strength is conservatively assumed to be 50,000 so the curve for a ratio of .7 should be used. (A ratio equal to or higher than the actual ratio must be used for conservatism.) For a 9-in. thick vessel ($\sqrt{t} = 3$), the value of M_m from Figure G-2214.1 is 2.94. The K_I for this case is then:

$$K_I = (M_m)(\text{Membrane Stress})$$

$K_I = (2.94)(33,400) = 98,\cancel{3}200$[55] psi $\sqrt{\text{in}}$.

From Figure G-2210.1 (reproduced here as Fig. 2), a temperature of at least RT_{NDT} + 120°F is necessary for a K_I of this level.

If, for example, an original RT_{NDT} of 40°F is assumed, the required temperature is then 40 + 120, or 160°F.

b. Inservice Leak and Hydrotest.

The temperatures for the inservice leak and hydrotest, performed at operating pressure and about 1.1 times the[56] operating pressure, respectively, are calculated in essentially the same way. The differences are that a factor of 1.5 must be applied to the calculated K_I to provide extra margin, and the stress levels are lower, so the value of M_m is taken from a lower ratio curve.

Using the same vessel as an example, with a normal operating pressure (P_0) of 2250 psi, the membrane stress for the leak test can be approximated as:

$$\frac{\text{operating pressure x allowable stress}}{\text{design pressure}}$$

or $\frac{2250 \times 26,700}{2500} = 24,000$ psi

This is about half of the minimum yield strength, so the M_m is taken from the 0.5 ratio curve, and is 2.87. The calculated K_I that must be assumed is then:

$$K_I = (1.5)\ (M_m)(\text{Membrane Stress})$$

$$\text{or } K_I = (1.5)(2.87)(24{,}000) = 103{,}\cancel{5}300\text{[57]}\ \text{psi}\ \sqrt{\text{in}}.$$

From the $K_{I\cancel{R}a}$[58] curve, a temperature of about RT_{NDT} + 125°F is required. As this is an inservice test, the RT_{NDT} would probably have been increased from its original value of + 40°F by some shift caused by radiation. Assume this shift is 100°F, thus the temperature for the leak test must be at least:

$$40 + 100 + 125 = 265°\text{F}$$

The inservice hydrotest temperature (at 1.1 P_0) is determined in exactly the same way, and requires a minimum temperature of about RT_{NDT} + 133°F, or 273°F.

c. Heatup, Cooldown, and Normal Operation.

For normal operation, which includes upset conditions and startup and shutdown procedures, operating limit curves must be provided that show the maximum permissible pressure at any temperature from cold shutdown conditions to full pressurization conditions.

Reactor vendors have developed computer codes to perform the necessary calculations, because thermal stresses must be included, and hand calculations of even moderate sophistication are very time consuming. WRC Bulletin 175 includes a set of curves derived from computer programs that can be used to approximate the K_I caused by thermal stresses, as a function of wall thickness and rate of temperature change. Pressure-temperature curves developed using these approximations agree fairly well with those determined using much more rigorous procedures, and can be used with confidence to evaluate the proposed operating limits given in Technical Specifications. These curves require the calculation of only 3 to 5 points. Either allowable pressure at a given temperature, or allowable temperature at a given pressure can be calculated. It is usually more convenient to calculate allowable minimum temperature, so this method will be used in the example.

Using the same reactor vessel as in the previous example, and a rate of temperature change of 50°F per hour, calculations of required temperatures for several pressures are illustrated. The curves for thermal effects given in WRC Bulletin 175 are very conservative, thus no additional margin need ~~by~~be[59] applied to the K_I from thermal stress, but a factor of 2.0 is used on primary stresses. The basic expression is then:

$$K_{I\cancel{R}a}\text{[60]} \geq 2\ K_I(\text{membrane}) + K_I\ (\text{thermal})$$

K_I(membrane) is calculated exactly as in the previous examples. K_I(thermal) for a 9-in. thick wall, at 50°F/hr is about 12,000 psi √in. from Figure 4-5, WRC Bulletin 175 (reproduced here as Fig. 3).

Thus, for a pressure of 2250 psig, a membrane stress of 24,000 psi, and M_m of 2.87, the basic expression is given by

$$K_{I\cancel{R}a}{}^{61} > (2)(24{,}000)(2.87) + 12{,}000 = 150{,}000 \text{ psi } \sqrt{\text{in.}}$$

From the $K_{I\cancel{R}a}$[62] curve, a temperature of RT_{NDT} + 158°F is required. With an RT_{NDT} of 140°F (including irradiation effects), the temperature required for operating pressure at a heatup or cooldown rate of 50°F/hr is then

$$140 + 158 = 298°F$$

For a pressure of 1/2 of operating (1125 psig), the membrane stress is 1/2 of that at operating pressure, or 12,000 psi.

The M_m can be taken from the 0.5 σ/σ_y ratio curve in Figure G-2214.1 (reproduced as Figure 1 herein), so is again 2.87.

$$K_{I\cancel{R}a}{}^{63} \geq (2)(12{,}000)(2.87) + 12{,}000 = 81{,}000 \text{ psi } \sqrt{\text{in.}}$$

From the $K_{I\cancel{R}a}$[64] curve, the minimum temperature is RT_{NDT} + 100°F, or 140 + 100 = 240°F.

The same calculation for a pressure of 1/5 operating pressure (450 psig and 4800 psi stress) is similar, but in this case the stress is less than .1 of the yield strength, so the M_m (from the .1 ratio curve) is only 2.82.

$$K_{I\cancel{R}a}{}^{65} \geq (2)(4800)(2.82) + 12{,}000 = 39{,}000 \text{ psi } \sqrt{\text{in.}}$$

The $K_{I\cancel{R}a}$[66] curve shows that the minimum temperature is RT_{NDT} + 0°F, or 140°F.

Three points on a 50°F/hr operating limit curve for this vessel at this time in its service lifetime have thus been calculated:

Pressure (psig)	Min. Temperature (Fahrenheit)
450	140
11~~50~~25[67]	240
2250	298

The difference between water and metal temperatures at the 1/4T and 3/4T locations should be included in construction of the pressure-temperature limit

curves.[68] A smooth curve drawn through these points will very closely approximate the results using more rigorous methods.

d. Core Operation

Appendix G, 10 CFR Part 50, specifies pressure-temperature limits for core operation to provide additional margin during actual power production.

The pressure-temperature limits for core operation (except for low power physics tests) are that the reactor vessel must be at a temperature equal to or higher than the minimum temperature required for the inservice hydrostatic test, and at least 40°F higher than the minimum pressure-temperature curve for heatup and cooldown calculated as described in the preceding section. The minimum temperature for the inservice hydrostatic test for the vessel used in the preceding example was 273°F. A vertical line at 273°F on the pressure-temperature curve, intersecting a curve 40°F higher than the pressure-temperature limit curve as determined in the preceding section, constitutes the limit for core operation for this example.

4. <u>Pressurized Thermal Shock in PWRs</u>

The reviewer evaluates the projected values for RT_{PTS}, including the calculational methods and assumptions, and compares the projected values with the screening criteria in 10 CFR 50.61. For each PWR where the RT_{PTS} value for any material in the beltline is projected to exceed the PTS screening criterion before the expiration date of the operating license, the licensee shall submit an analysis and schedule for implementation of flux reduction programs that are reasonably practical to avoid exceeding the PTS screening criterion. If the analysis indicates that no reasonably practical flux reduction program will prevent the value of RT_{PTS} from exceeding the PTS screening criterion before the expirations date of the operating license, the licensee shall submit a safety analysis to determine what modifications are necessary to prevent potential failure of the reactor vessel as a result of postulated PTS events if continued operation beyond the screening criterion is allowed. These safety analyses are reviewed against the requirements of 10 CFR 50.61 and the guidance of Regulatory Guide 1.154.[69]

For standard design certification reviews under 10 CFR 52, the procedures above should be followed, as modified by the procedures in SRP Section 14.3 (proposed), to verify that the design set forth in the standard safety analysis report, including inspections, tests, analysis, and acceptance criteria (ITAAC), site interface requirements and combined license action items, meet the acceptance criteria given in subsection II. SRP section 14.3 (proposed) contains procedures for the review of certified design material (CDM) for the standard design, including the site parameters, interface criteria, and ITAAC.[70]

IV. EVALUATION FINDINGS

The reviewer verifies that sufficient information has been provided to satisfy the requirements of this SRP section and that the completeness and technical adequacy of ~~his~~the[71] evaluation will support the following statement in the staff's safety evaluation report:

> The pressure-temperature limits imposed on the reactor coolant system for all operating and testing conditions to assure adequate safety margins against nonductile or rapidly propagating failure are in conformance with the fracture toughness criteria of Appendix G of 10 CFR Part 50 and Section III, including Appendix G, "Protection Against Nonductile Failure," of the ASME Boiling and Pressure Vessel Code. The change in fracture toughness requirements of the pressure vessel during operation will be determined by Appendix H of 10 CFR Part 50. Thermal shock events have been adequately addressed in accordance with 10 CFR 50.61 (PWRs only).[72] The use of operating limits, based upon the criteria defined in Standard Review Plan Section 5.3.2, provides reasonable assurance that non-ductile or rapidly propagating failure will not occur, and constitutes an acceptable basis for satisfying the requirements of ~~paragraph~~Sections[73] 50.55a, 50.60,[74] and 50.61 (PWRs only)[75] of 10 CFR Part 50 and General Design Criteria 1, 14, 31 and 32 of Appendix A of 10 CFR Part 50.

For design certification reviews, the findings will also summarize, to the extent that the review is not discussed in other safety evaluation report sections, the staff's evaluation of inspections, tests, analyses, and acceptance criteria (ITAAC), including design acceptance criteria (DAC), site interface requirements, and combined license action items that are relevant to this SRP section.[76]

V. IMPLEMENTATION

The following is intended to provide guidance to applicants and licensees regarding the NRC staff's plan ~~to~~for[77] using this SRP section.

This SRP section will be used by the staff when performing safety evaluations of license applications submitted by applicants pursuant to 10 CFR 50 or 10 CFR 52.[78] Except in those cases in which the applicant proposes an acceptable alternative method for complying with specified portions of the Commission's regulations, the method described herein will be used by the staff in its evaluation of conformance with Commission regulations.

The provisions of this SRP section apply to reviews of applications docketed six months or more after the date of issuance of this SRP section.[79]

Implementation schedules for conformance to parts of the method discussed herein are contained in the referenced regulations and regulatory guides.[80]

VI. REFERENCES

~~8~~1.[81] 10 CFR ~~Part 50, paragraph~~ [82]50.55a, "Codes and Standards."

2. 10 CFR 50.60, "Acceptance Criteria for Fracture Prevention Measures for Lightwater Nuclear Power Reactors for Normal Operation."[83]

3. 10 CFR 50.61, "Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events."[84]

~~1~~4. 10 CFR Part 50, Appendix A, General Design Criteria 1~~, 14, 31, and 32~~ "Quality Standards and Records."[85]

5. 10 CFR Part 50, Appendix A, General Design Criterion 14, "Reactor Coolant Pressure Boundary."

6. 10 CFR Part 50, Appendix A, General Design Criterion 31, "Fracture Prevention of Reactor Coolant Pressure Boundary."

7. 10 CFR Part 50, Appendix A, General Design Criterion 32, "Inspection of Reactor Coolant Pressure Boundary."

~~2~~8. 10 CFR Part 50, Appendix G, "Fracture Toughness Requirements."

~~3~~9. 10 CFR Part 50, Appendix H, "Reactor Vessel Material Surveillance Program Requirements."

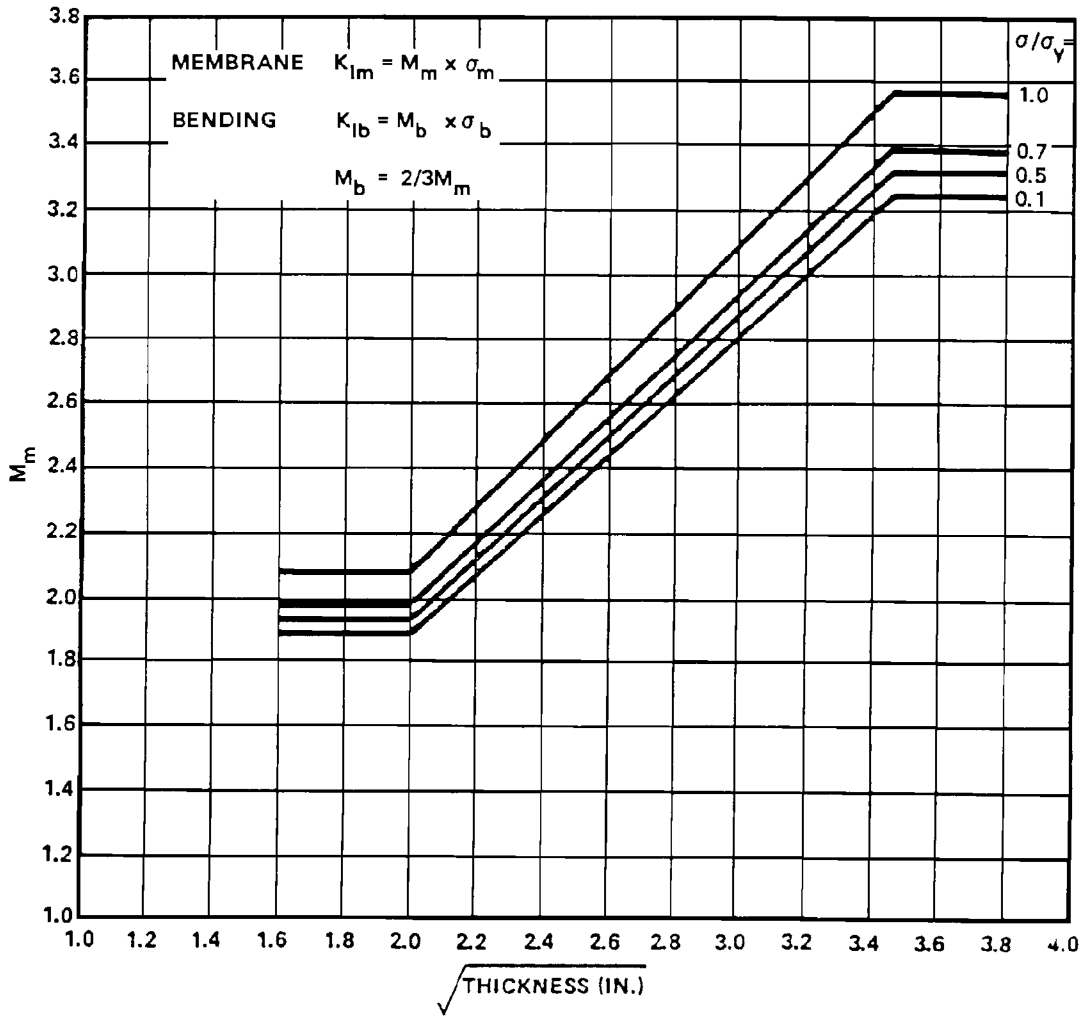
~~7~~10. Regulatory Guide 1.99, "~~Effects of Residual Elements on Predicted~~ Radiation Embrittlement of~~Damage to~~[86] Reactor Vessel Materials."

11. Regulatory Guide 1.154, "Format and Content of Plant-Specific Pressurized Thermal Shock Safety Analysis Reports for Pressurized Water Reactors."[87]

~~6~~12. Branch Technical Position - ~~MTEB~~ EMCB[88] 5-2, "Fracture Toughness Requirements~~ for Older Plants~~[89]," attached to this SRP section.

~~4~~13. ASME Boiler and Pressure Vessel Code, Section III, including Appendix G, "Protection Against Nonductile Failure," American Society of Mechanical Engineers.

~~5~~14. WRC Bulletin 175, "PVRC Recommendation on Fracture Toughness," Welding Research Council, Pressure Vessel Research Committee Ad Hoc Group on Toughness Requirements, August 1972.[90]



M_m AND M_b VS. WALL THICKNESS FOR

SEMI-ELLIPTICAL SURFACE FLAW ¼T DEEP AND 1½T LONG

FIGURE 1

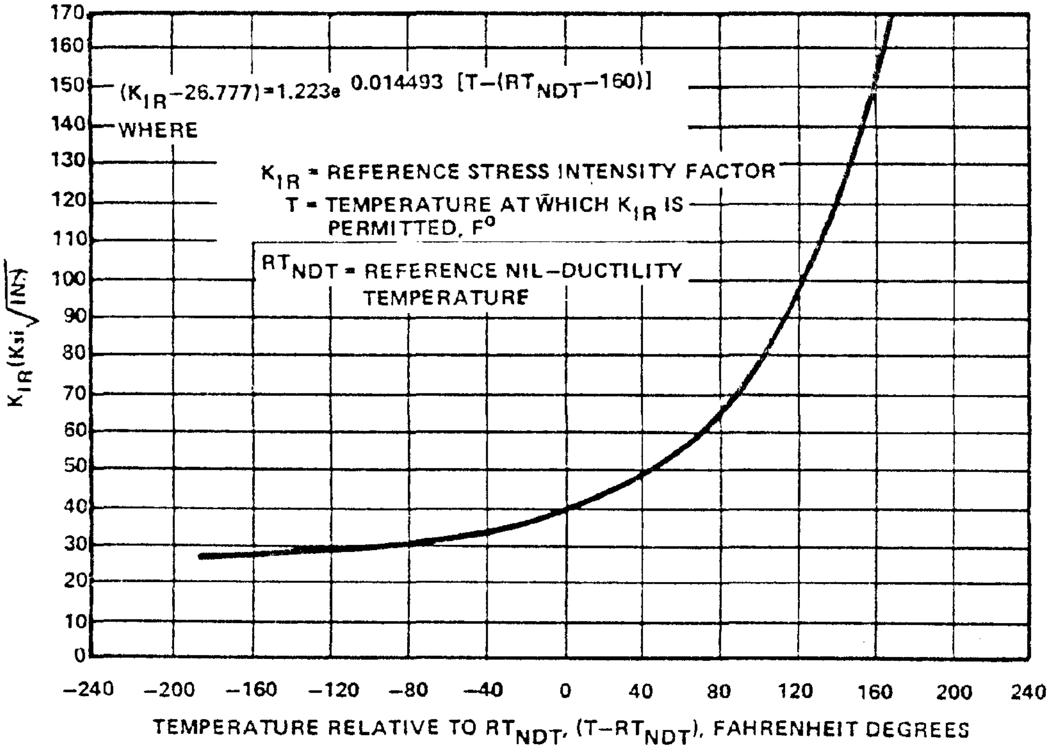
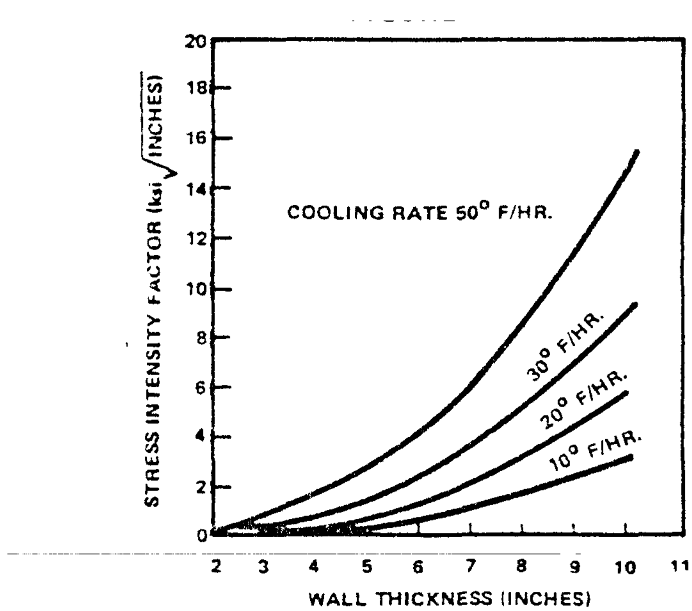


FIGURE 2



4-5 – STRESS INTENSITY FACTOR CAUSED BY THERMAL STRESS FOR CYLINDERS WITH RADIUS/THICKNESS = 10

FIGURE 3

BRANCH TECHNICAL POSITION - ~~MTEB~~ EMCB 5-2
(Formerly MTEB 5-2)[91]
FRACTURE TOUGHNESS REQUIREMENTS

A. Background

Current requirements regarding fracture toughness, pressure-temperature limits, and material surveillance are covered by the ASME Code and Appendices A, G, and H to 10 CFR Part 50. The purpose of this branch technical position is to summarize these requirements and provide clarification, as necessary.

Since many of these requirements were not in force when some plants were designed and built, this position also provides guidance for applying these requirements to older plants. Also included is a description of acceptable procedures for making the conservative estimates and assumptions for older plants that may be used to show compliance with the new requirements. It should be noted that the applicants must present adequate technical justifications for any estimates of material properties required by the regulations before exemption to the regulations may be granted.

B. Branch Technical Position

1. Preservice Fracture Toughness Test Requirements.

The fracture toughness of all ferritic materials used for pressure-retaining components of the reactor coolant pressure boundary shall be evaluated in accordance with the requirements of Appendix G, 10 CFR Part 50, as augmented by Section III of the ASME Code. The fracture toughness test requirements for plants with construction permits prior to August 15, 1973 may not comply with the new Codes and Regulations in all respects. The fracture toughness of the materials for these plants must be assessed by using the available test data to estimate the fracture toughness in the same terms as the new requirements. This must be done because the operating limitations imposed on old plants must provide the same safety margins as are required for new plants.

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[92] results of fracture toughness tests. Both drop weight nil-ductility transition temperature (NDTT)[93] tests and Charpy V-notch tests must be run to determine the RT_{NDT}. The NDTT temperature, as determined by drop weight tests (ASTM E-208-1969[94])[95] is the RT_{NDT} if, at 33°C (60°F)[96] above the NDTT, at least 68 J (50 ft-lbs,[97] of energy and 0.89 mm (35 mils)[98] lateral expansion (LE)[99] are obtained in Charpy V-notch[100] tests on specimens oriented in the weak direction (transverse[101] 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 required 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[102].

(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)[103] was obtained in Charpy V-notch tests, or -18°C (0°F)[104], 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)[105].

(b) The temperatures of the Charpy V-notch upper shelf.

(c) The temperature at which 136 J (100 ft-lbs)[106] was obtained on Charpy V-notch tests if the upper-shelf energy values were above 136 J (100 ft-lbs)[107].

(3) If transversely-oriented Charpy V-notch specimens were not tested, the temperature at which 68 J (50 ft-lbs)[108] and 0.89 mm (35 mils)[109] LE would have been obtained on transverse[110] 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)[111] and 0.89 mm (35 mils)[112] LE were obtained on longitudinally-oriented specimens increased 11°C (20°F)[113] to provide a conservative estimate of the temperature that would have been required 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)[114] was obtained, that temperature may be used as an estimate of the RT_{NDT} provided that at least 61 J (45 ft-lbs)[115] was obtained if the specimens were longitudinally oriented. If the minimum value obtained was less than 61 J (45 ft-lbs)[116], the RT_{NDT} may be estimated as 11°C (20°F)[117] above the test temperature.

1.2 Estimation of Charpy V Upper-Shelf Energies

For the beltline region of reactor vessels, the upper shelf toughness must be adequate to accommodate degradation by neutron radiation. The original minimum shelf energy must be 102 J (75 ft-lbs)[118] for vessels with an estimated end-of-life[119] neutron fluence (> 1 MeV) of $1x10^{19}$ and over. A value of 95 J (70 ft-lbs)[120] is considered adequate for material for vessels that will be subjected to lower fluences.

If upper-shelf Charpy energy values were not obtained, conservative estimates should be made using results of tests on specimens from the first surveillance capsule removed.

If tests were only made on longitudinal specimens, the values should be reduced to 65% of the longitudinal values to estimate the transverse properties.

The predicted end-of-life Charpy upper shelf energy and adjusted reference temperature for the reactor vessel materials must meet the requirements of 10 CFR 50, Appendix G, paragraph IV.B. Reactor vessel materials that do not meet the specified end-of-life acceptance criteria are reviewed in accordance with paragraphs V.C and V.D of 10 CFR 50, Appendix G. NUREG-0744 provides an acceptable methodology for performance of fracture analysis for demonstrating adequate margins of safety for continued operation in accordance with 10 CFR 50, Appendix G, paragraph V.C.3.[121]

1.3 Reporting Requirements

Fracture toughness information required by the Code and by Appendix G, 10 CFR Part 50, must be reported in the FSAR to provide a basis for evaluating the adequacy of the operating limitations given in the Technical Specifications. In the case of older plants, the data may be estimated, using the procedures listed above, or other methods that can be shown to be conservative.

2. Operating Limitations for Fracture Toughness

2.1 Required Pressure-Temperature Operating Limitations

As required by Appendix G, 10 CFR Part 50, the following operating limitations shall be determined and included in the Technical Specifications. The basis for determination shall be reported, and is the responsibility of the applicant, but in no case shall the limitations provide less safety margin than those determined in accordance with Appendix G, 10 CFR Part 50, and Appendix G to Section III of the Code.

(1) Minimum temperatures for performing any hydrostatic test involving pressurization of the reactor vessel after installation in the system.

(2) Minimum temperatures for all leak and hydrostatic tests performed after the plant is in service.

(3) Maximum pressure-minimum temperature curves for operation, including startup, upset, and cooldown conditions.

(4) Maximum pressure-minimum temperature curves for core operation.

2.2 Recommended Bases for Operating Limitations

2.2.1 Leak and Hydrostatic Tests

(1) It is recommended that no tests at pressures higher than design pressure be conducted with fuel in the vessel.

(2) Tests at pressures less than design pressure should be conducted at temperatures calculated according to Appendix G of Section III of the Code for the beltline region (including conservative estimates of radiation damage, see Section 3.0 below) if the maximum calculated primary stress in no other region of the vessel exceeds 1.25 S_m during the test, and the RT_{NDT} of the beltline is assumed to be at least 17°C (30°F)[122] above that of the higher stressed regions. If primary stresses are calculated to be over 1.25 S_m in any region during the test, the RT_{NDT} of the vessel must be assumed to be at least 28°C (50°F)[123] higher than that of any region where the calculated primary stresses are over 1.25 S_m.

(3) Alternatively, a fracture mechanics analysis, with technical justification for all assumptions and bases, may be made to determine the minimum test temperature. In no event shall the minimum temperature be lower than that resulting from calculations for the beltline region in accordance with Appendix G of the Code.

2.2.2 Heatup and Cooldown Limit Curves

Heatup and cooldown pressure-temperature limit curves may be determined using single pr/t stress calculations, using the method given in Appendix G of the Code. The effect of thermal gradients may be conservatively approximated by the procedures in Appendix G of the Code or from Figure 4-5 in WRC Bulletin 175.

Calculations need only be performed for the beltline region, if the RT_{NDT} of the beltline is demonstrated to be adequately higher than the RT_{NDT} for all higher stressed regions.

Alternatively, more rigorous analytical procedures may be used, provided that the intent of the Code is met, and adequate technical justification for all assumptions and bases is provided.

2.2.3 Core Operation Limits

To provide added margins during actual core operation, Appendix G, 10 CFR Part 50 requires a minimum temperature during core operation, and a 22°C (40°F)[124] margin in temperature over the pressure-temperature limits as determined for heatup and cooldown in 2.2.2 above. The minimum temperature, regardless of pressure, is the temperature calculated for the inservice hydrostatic test according to 2.2.1 above.

2.2.4 Upset Conditions

The pressure-temperature limits described in 2.2.2 and 2.2.3 above are applicable to upset conditions. Normal operating procedures must permit variations from intended operation, including all upset conditions, without exceeding the limit curves.

2.2.5 Emergency and Faulted Conditions

It is recognized that the severity of a transient resulting from an emergency or faulted condition is not directly related to operating conditions, and resulting temperature-stress relationships in the reactor coolant boundary components are primarily system dependent, and therefore not under direct control of the operator.

For these reasons, operating limits for emergency and faulted conditions are not a requirement of the Technical Specifications.

The SAR should present descriptions of the continued integrity of all vital components of the RCPB during postulated faulted conditions. It is recommended that such descriptions be made in as realistic a manner as possible, avoiding grossly overconservative assumptions and procedures.

2.3 Reporting Requirements

The Technical Specifications must include the operating and test limits discussed above, and the basis for their determination. The Technical Specifications must also include information on the intended operating procedures, and justify that adequate margins between the expected conditions and the limit conditions will be provided to protect against unexpected or upset conditions.

3. Inservice Surveillance of Fracture Toughness

The reactor vessel may be exposed to significant neutron radiation during the service life. This will affect both the tensile and toughness properties. A material surveillance program in conformance with Appendix H, 10 CFR Part 50, must be carried out.

3.1 Surveillance Program Requirements

The minimum requirements for the surveillance program are covered by Appendix H, 10 CFR Part 50. It is strongly recommended that consideration ~~by~~be[125] given to the desirability of additional surveillance methods, such as the inclusion of CT, DWT, DT, or other specimens to provide the capability of redundant test methods and analytical procedures, particularly if the estimated neutron fluence is over $2x10^{19}$, or the toughness of the vessel material is marginal.

The selection of material to be included in the surveillance program should be in accordance with ASTM E-185-~~73~~82[126], unless the intent of the program is better realized by using more rigorous criteria. For example, the approach of estimating the actual RT_{NDT} and upper shelf toughness of each plate, forging, or weld in the beltline as a function of service life, and choosing as the surveillance materials those that are expected to be most limiting, may be preferable in some cases. This would include consideration of the initial RT_{NDT}, the upper shelf toughness, the expected radiation sensitivity of the material (based on copper and ~~phosphorous~~nickel[127] content, for example) and the neutron fluence expected at its location in the vessel.

3.2 SAR Requirements

The adequacy of the surveillance program cannot be evaluated unless all pertinent information is included in the SAR. Information requested for beltline materials includes the following:

(1) Tensile properties.

(2) DWT and Charpy V test results used to determine RT_{NDT}.

(3) Charpy V test results to determine the upper shelf toughness.

(4) Composition, specifically the copper and phosphorous content.

(5) Estimated maximum fluence for each beltline material.

(6) List of materials included in the surveillance program, with basis used for their selection.

3.3 Surveillance Test Procedures

Surveillance capsules must be removed and tested at intervals in accordance with Appendix H, 10 CFR Part 50. The proposed removal and test schedule shall be included in the Technical Specifications.

3.4 Reporting Requirements

All information used to evaluate results of the tests on surveillance materials, evaluation methods, and results of the evaluation should be submitted with the evaluation report. This should include:

(1) Original properties and compositions of the materials.

(2) Fluence calculations, including original predictions, for both surveillance specimens and vessel wall.

(3) Test results on surveillance specimens.

(4) Basis for evaluation of changes in RT_{NDT} and upper shelf toughness.

(5) Updated prediction of vessel properties.

3.5 Technical Specification Changes

Changes in the operating and test limits recommended as a result of evaluating the properties of the surveillance material, together with the basis for these changes, shall be submitted to the Division of ~~Licensing~~Operating Reactor Support[128] for approval.

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SRP Draft Section 5.3.2

Attachment A - Proposed Changes in Order of Occurrence

Item numbers in the following table correspond to superscript numbers in the redline/strikeout copy of the draft SRP section.

Item	Source	Description
1.	**Integrated Impact 972,** PRB Comment	Revised the SRP section title to include "pressurized thermal shock."
2.	Current PRB names and abbreviations.	Editorial change made to reflect current PRB names and responsibilities for SRP Section 5.3.2.
3.	Editorial.	10 CFR 50.55a is more appropriately referred to as a "Section," according to the convention for citing portions of the CFR.
4.	Editorial.	To be consistent with the remainder of the section the acronym SSC was identified for structures, systems, and components.
5.	Editorial, PRB Comment	Added hyphen to "non-brittle."
6.	**Integrated Impact 972.**	Added an Area of Review to incorporate 10 CFR 50.61 regarding pressurized thermal shock (PTS).
7.	Editorial, **PI # 25117**	The text was revised to clarify the applicability of pressure-temperature limits and to be consistent with the requirements of 10 CFR 50, Appendix G.
8.	SRP-UDP format item, Review Interfaces, Editorial.	Added Review Interface subsection per SRP-UDP guidance. Although the proposed interfaces are not included in existing SRP Section 5.3.2, this is considered an editorial change. The review of pressure-temperature limits per SRP Section 5.3.2 is a subset of the review conducted per SRP Sections 5.2.3 and 5.3.1 for RCPB and reactor vessel materials and the compliance of those materials with the fracture toughness requirements of 10 CFR 50 Appendix G. The proposed review interface with SRP Section 5.2.2 is appropriate since the over-pressure-protection system must be designed to ensure that pressure-temperature limits established in SRP Section 5.3.2 are not exceeded under conditions of normal and low temperature operations, and anticipated overpressure events.
9.	Editorial, **PI # 23464**	Added a review interface with the SRXB and SRP Section 4.3 with regard to review of the reactor vessel wall fluence. The neutron fluence is used in calculations for adjusting the reference temperature limits to account for irradiation effects on material fracture toughness.

Item	Source	Description
10.	Editorial	The Acceptance Criteria description was modified from a paragraph style discussion to a alpha-numeric listing. The purpose of this change was to (1) create format consistency with other SRP sections, (2) to provide a clear and concise listing of the general acceptance criteria, (3) develop a format that is more flexible with regard to adding additional (new) acceptance criteria, and (4) to facilitate separation of the general acceptance criteria provided by the Commission's regulations from the more specific criteria associated with implementing those regulations. Some documents such as Regulatory Guide 1.99, the ASME Code, and WRC Bulletin 175 were moved from the discussion in the acceptance criteria listing to a specific criteria subsection consistent with other SRP sections. 10 CFR 50, Appendix H was deleted from the Acceptance Criteria since compliance with the requirements of Appendix H are reviewed under SRP Section 5.3.1.
11.	**Integrated Impact 446.**	Added 10 CFR 50.60, regarding implementation of 10 CFR 50 Appendices G and H, to the acceptance criteria.
12.	**Integrated Impact 972.**	Added 10 CFR 50.61, regarding pressurized thermal shock, to the acceptance criteria.
13.	Editorial	Added a typical lead-in sentence for specific acceptance criteria. This change in conjunction with the reformatting of the individual acceptance criteria, separates the more detailed criteria from the more general acceptance criteria and is consistent with other SRP sections.
14.	**Integrated Impact 446.**	Added 10 CFR 50.60 to discussion of applicable regulations because of its relationship to 10 CFR 50, Appendices G and H
15.	Editorial	Added comma to accommodate changes to sentence structure as implemented by Integrated Impact 446.
16.	SRP-UDP format item, Reformat References	Added parenthetical identification to ASME Code.
17.	**Integrated Impact 972.**	Added a discussion of 10 CFR 50.61 and Regulatory Guide 1.154 in specific criteria II.1 related to applicable regulations.
18.	Current PRB names and abbreviations.	Editorial change made to reflect current PRB names and responsibilities for BTP MTEB 5-2.
19.	**Integrated Impact 446.**	Added reference to 10 CFR 50.60 and Appendix G in support of the existing statements in the SRP.

Item	Source	Description
20.	SRP-UDP Integration, PRB Comment	Changed K_{IR} to K_{Ia} in accordance with PRB Comments.
21.	SRP-UDP Integration, PRB Comment	Changed K_{IR} to K_{Ia} in accordance with PRB Comments.
22.	SRP-UDP Integration, PRB Comment	Changed K_{IR} to K_{Ia} in accordance with PRB Comments.
23.	**Integrated Impact 445,** Reference Verification	The revision to Regulatory Guide 1.99 indicates that copper and nickel are of greater importance than phosphorous, and phosphorous is no longer considered in the chemistry factor as applied in determining the adjusted RT_{NDT} per the Regulatory Guide.
24.	**Integrated Impact 445,** Reference Verification	Revised the text to indicate that Regulatory Guide 1.99 provides methods for determining the shift in reference temperature due to irradiation.
25.	Editorial	Changed citation of Welding Research Council document WRC Bulletin 175 to be consistent with the other citations throughout the SRP section.
26.	SRP-UDP format item, Reformat References	Added parenthetical identifier to the reference for WRC Bulletin 175.
27.	SRP-UDP Integration, PRB Comment	Changed K_{IR} to K_{Ia} in accordance with PRB Comments.
28.	SRP-UDP Integration, PRB Comment	Changed K_{IR} to K_{Ia} in accordance with PRB Comments.
29.	SRP-UDP Integration, PRB Comment	Changed K_{IR} to K_{Ia} in accordance with PRB Comments.
30.	SRP-UDP Integration, PRB Comment	Changed K_{IR} to K_{Ia} in accordance with PRB Comments.
31.	SRP-UDP Integration, PRB Comment	Changed K_{IR} to K_{Ia} in accordance with PRB Comments.
32.	SRP-UDP Integration, PRB Comment	Changed K_{IR} to K_{Ia} in accordance with PRB Comments.
33.	NRC Metrication Policy	Converted the cited value of 40°F to the metric equivalent of 22°C. Revised the text to cite the value in the dual unit format in accordance with the metrication policy and SRP-UDP guidance.
34.	**Integrated Impact 972,** NRC Metrication Policy Implementation	Added specific criteria II.3.e describing the acceptance criteria related to temperature limits for PTS considerations. Cited values of 270°F and 300°F from 10 CFR 50.61 were converted to metric equivalents of 132°C and 149°C.

SRP Draft Section 5.3.2

Attachment A - Proposed Changes in Order of Occurrence

Item	Source	Description
35.	SRP-UDP format item, Technical Rationale	Added Technical Rationale for GDCs 1, 14, 31, and 32, 10 CFR 50.55a, 10 CFR 50.60, 10 CFR 50.61, and 10 CFR 50, Appendix G.
36.	Editorial, **Integrated Impact 1131.**	Revised the subheading to accommodate the 10 CFR 52 licensing process with regard to review of pressure-temperature limits.
37.	Editorial, **Integrated Impact 1131.**	Revised the text to accommodate the 10 CFR 52 licensing process with regard to review of pressure-temperature limits at the construction permit or design certification stages..
38.	**Integrated Impact 446.**	Incorporated 10 CFR 50.60 into review procedures discussion for PSAR.
39.	**Integrated Impact 972.**	Incorporated 10 CFR 50.61 into the discussion of requirements applicable to the reactor vessel beltline that are reviewed as part of the PSAR.
40.	Editorial	Added text to indicate that Regulatory Guide 1.99 provides guidance and not requirements.
41.	SRP-UDP format item, Reformat References	Deleted title to Regulatory Guide 1.99 consistent with SRP-UDP guidance for formatting of references. The complete reference including title is provided in Subsection VI.
42.	Editorial, **Integrated Impact 1131.**	Revised the subheading to accommodate the 10 CFR 52 licensing process with regard to review of pressure-temperature limits.
43.	Current PRB names and abbreviations.	Editorial change made to reflect current PRB names and responsibilities for SRP Section 5.3.2.

Item	Source	Description
44.	NRC Metrication Policy	Metrication was not performed for the example calculations in the Review Procedures. The calculations are used to illustrate the methods to be employed by the reviewer and are not specific to any of the requirements or criteria in the section. There does not appear to be any value in converting these numbers/units to SI equivalents. Related to this issue is the figures provided with the Section. These figures were also not converted to metric equivalents. The figures are provided for the purpose of the example calculations and are not otherwise referred to in the SRP section. The figures are also taken directly from the ASME Code and an associated reference (i.e., Welding Research Council Bulletin 175). Review of these references confirmed the figures are provided in english units only. To convert the figures to metric equivalents would be inappropriate since the ASME Code is stated as the basis for the methods that utilize the figures. In addition, metric conversion of the figures could potentially introduce inaccuracies in the calculations.
45.	Editorial	Added the words "times the" to clarify and grammatically correct the sentence with regard to the multiplier applied to the design pressure.
46.	NRC Metrication Policy	Converted the cited value of RT_{NDT} + 60°F to the metric equivalent of RT_{NDT} + 33°C. Revised the text to cite the value in the dual unit format in accordance with the metrication policy and SRP-UDP guidance.
47.	Current PRB names and abbreviations.	Editorial change made to reflect current PRB names and responsibilities for SRP Section 5.3.2.
48.	Editorial	Added text to define the existing acronym for NDTT.
49.	Current PRB names and abbreviations.	Editorial change made to reflect current PRB names and responsibilities for SRP Section 5.3.2.
50.	SRP-UDP Integration, PRB Comment	Changed K_{IR} to K_{Ia} in accordance with PRB Comments.
51.	Current PRB names and abbreviations.	Editorial change made to reflect current PRB names and responsibilities for SRP Section 5.3.2.
52.	SRP-UDP Integration, PRB Comment	Changed K_{IR} to K_{Ia} in accordance with PRB Comments.
53.	Editorial	Correct the calculated value to be 33,000 instead of 33,400.
54.	Editorial	The text was modified to account for the corrected differences in the calculated stress values by stating the more conservative value is used.

SRP Draft Section 5.3.2
Attachment A - Proposed Changes in Order of Occurrence

Item	Source	Description
55.	Editorial	Correct the calculated value to be 98,200 instead of 98,300.
56.	Editorial	Added the words "times the" to clarify and grammatically correct the sentence with regard to the multiplier applied to the operating pressure.
57.	Editorial	Correct the calculated value to be 103,300 instead of 103,500.
58.	SRP-UDP Integration, PRB Comment	Changed K_{IR} to K_{Ia} in accordance with PRB Comments.
59.	SRP Integration, PRB Comment	Corrected typographical error.
60.	SRP-UDP Integration, PRB Comment	Changed K_{IR} to K_{Ia} in accordance with PRB Comments.
61.	SRP-UDP Integration, PRB Comment	Changed K_{IR} to K_{Ia} in accordance with PRB Comments.
62.	SRP-UDP Integration, PRB Comment	Changed K_{IR} to K_{Ia} in accordance with PRB Comments.
63.	SRP-UDP Integration, PRB Comment	Changed K_{IR} to K_{Ia} in accordance with PRB Comments.
64.	SRP-UDP Integration, PRB Comment	Changed K_{IR} to K_{Ia} in accordance with PRB Comments.
65.	SRP-UDP Integration, PRB Comment	Changed K_{IR} to K_{Ia} in accordance with PRB Comments.
66.	SRP-UDP Integration, PRB Comment	Changed K_{IR} to K_{Ia} in accordance with PRB Comments.
67.	Editorial	Corrected typographical error. The reported values in the table are from the example calculations preceding the table. The calculations are performed for pressures of 2250, 1125, and 450 psig.
68.	PBR Comment	Added discussion with regard to use of temperature differences at the 1/4T and 3/4T locations in determining the pressure-temperature limit curves.
69.	**Integrated Impact 972.**	Incorporated the pressurized thermal shock rule, 10 CFR 50.61, and Regulatory Guide 1.154 into the Review Procedures for SRP Section 5.3.2.
70.	SRP integration format item	Added boiler-plate paragraph regarding reviews conducted for design certifications.
71.	Editorial	Changed the text to be gender neutral.
72.	**Integrated Impact 972.**	Incorporated conclusions relative to thermal shock in the Evaluation Findings for SRP Section 5.3.2.

SRP Draft Section 5.3.2
Attachment A - Proposed Changes in Order of Occurrence

Item	Source	Description
73.	Editorial	The word "paragraph" was changed to "Sections" to be consistent with proper CFR citation convention. Sections was pluralized to accommodate the addition of 10 CFR 50.60, and 10 CFR 50.61.
74.	**Integrated Impact 446.**	Incorporated 10 CFR 50.60 into the Evaluation Findings.
75.	**Integrated Impacts 972.**	Incorporated 10 CFR 50.61 into the Evaluation Findings.
76.	10 CFR 52 Applicability	Added standard paragraph related to findings associated with review of applications in accordance with 10 CFR 52.
77.	Editorial	Revised text to improve clarity and grammer, and to be consistent with other SRP Sections.
78.	SRP-UDP Format Item	Added boiler-plate statements to the Implementation subsection to incorporate 10 CFR 52.
79.	SRP-UDP Format Item	Added boiler-plate statements to the Implementation subsection to address the general applicability of the SRP section to new and pending license applications.
80.	Editorial	The implementation subsection was revised to incorporate regulations in addition to Regulatory Guides, as providing implementation schedules.
81.	SRP-UDP Format Item, Reformat References	Reordered and renumbered references in accordance with SRP-UDP guidance.
82.	SRP-UDP format item, Reformat References	Revised the reference to 10 CFR 50.55a to delete the "paragraph" designation, which is inconsistent with CFR citation procedure, and reformat the reference according to SRP-UDP guidance.
83.	**Integrated Impact 446.**	Added reference to 10 CFR 50.60, which was incorporated into the SRP section.
84.	**Integrated Impact 972.**	Added reference to 10 CFR 50.61, which was incorporated into the SRP section.
85.	SRP-UDP format item, Reformat References, Editorial	Divided existing reference no. 1 describing all GDCs into separate references for each GDC and added the GDC titles.
86.	SRP-UDP format item, Reference Verification	Revised the title of Regulatory Guide 1.99 to be consistent with the latest revision (Revision 2).
87.	**Integrated Impact 972.**	Added reference to Regulatory Guide 1.154.
88.	Current PRB names and abbreviations.	Editorial change made to reflect current PRB names and responsibilities for BTP MTEB 5-2.

Item	Source	Description
89.	SRP-UDP format item, Reference Verification	The title for BTP MTEB 5-2 was corrected.
90.	SRP-UDP format item, Reference Verification	Added author and date information to the existing reference for Welding Research Council (WRC) Bulletin 175.
91.	Current PRB names and abbreviations.	Editorial change made to reflect current PRB names and responsibilities for BTP MTEB 5-2.
92.	Editorial	Grammatical corrections.
93.	Editorial	Identified the first usage of the acronym "NDTT" in the Branch Technical Position.
94.	**Integrated Impact 1378**	Revised the non-date-specific standard citation of ASTM E208 to add the date of the standard version in effect at the time the existing SRP section was published.
95.	**Integrated Impact 658,** SRP-UDP standards citation update	Consideration should be given to updating the citation of ASTM E208 pending the review and approval of the associated standard comparison.
96.	NRC Metrication Policy	Converted the cited value of 60°F to the metric equivalent of 33°C. Revised the text to cite the value in the dual unit format in accordance with the metrication policy and SRP-UDP guidance.
97.	NRC Metrication Policy	Converted the cited value of 50 ft-lbs to the metric equivalent of 68 J. Revised the text to cite the value in the dual unit format in accordance with the metrication policy and SRP-UDP guidance.
98.	NRC Metrication Policy	Converted the cited value of 35 mils to the metric equivalent of 0.89 mm. Revised the text to cite the value in the dual unit format in accordance with the metrication policy and SRP-UDP guidance.
99.	Editorial	Added the acronym "LE" for the term "lateral expansion." This acronym is used in the existing text but was not previously defined.
100.	Editorial, PRB Comment	Added "-notch" to properly indicate the type of test.
101.	Editorial, PRB Comment	Revised "traverse" to be "transverse."
102.	Editorial	Added text clarifying the location and applicability of the guidance provided in the BTP.
103.	NRC Metrication Policy	Converted the cited value of 30 ft-lbs to the metric equivalent of 41 J. Revised the text to cite the value in the dual unit format in accordance with the metrication policy and SRP-UDP guidance.

SRP Draft Section 5.3.2
Attachment A - Proposed Changes in Order of Occurrence

Item	Source	Description
104.	NRC Metrication Policy	Converted the cited value of 0°F to the metric equivalent of -18°C. Revised the text to cite the value in the dual unit format in accordance with the metrication policy and SRP-UDP guidance.
105.	NRC Metrication Policy	Converted the cited value of 60°F to the metric equivalent of 33°C. Revised the text to cite the value in the dual unit format in accordance with the metrication policy and SRP-UDP guidance.
106.	NRC Metrication Policy	Converted the cited value of 100 ft-lbs to the metric equivalent of 136 J. Revised the text to cite the value in the dual unit format in accordance with the metrication policy and SRP-UDP guidance.
107.	NRC Metrication Policy	Converted the cited value of 100 ft-lbs to the metric equivalent of 136 J. Revised the text to cite the value in the dual unit format in accordance with the metrication policy and SRP-UDP guidance.
108.	NRC Metrication Policy	Converted the cited value of 50 ft-lbs to the metric equivalent of 68 J. Revised the text to cite the value in the dual unit format in accordance with the metrication policy and SRP-UDP guidance.
109.	NRC Metrication Policy	Converted the cited value of 35 mils to the metric equivalent of 0.89 mm. Revised the text to cite the value in the dual unit format in accordance with the metrication policy and SRP-UDP guidance.
110.	Editorial, PRB Comment	Revised "traverse" to be "transverse."
111.	NRC Metrication Policy	Converted the cited value of 50 ft-lbs to the metric equivalent of 68 J. Revised the text to cite the value in the dual unit format in accordance with the metrication policy and SRP-UDP guidance.
112.	NRC Metrication Policy	Converted the cited value of 35 mils to the metric equivalent of 0.89 mm. Revised the text to cite the value in the dual unit format in accordance with the metrication policy and SRP-UDP guidance.
113.	NRC Metrication Policy	Converted the cited value of 20°F to the metric equivalent of 11°C. Revised the text to cite the value in the dual unit format in accordance with the metrication policy and SRP-UDP guidance.
114.	NRC Metrication Policy	Converted the cited value of 30 ft-lbs to the metric equivalent of 41 J. Revised the text to cite the value in the dual unit format in accordance with the metrication policy and SRP-UDP guidance.

Item	Source	Description
115.	NRC Metrication Policy	Converted the cited value of 45 ft-lbs to the metric equivalent of 61 J. Revised the text to cite the value in the dual unit format in accordance with the metrication policy and SRP-UDP guidance.
116.	NRC Metrication Policy	Converted the cited value of 45 ft-lbs to the metric equivalent of 61 J. Revised the text to cite the value in the dual unit format in accordance with the metrication policy and SRP-UDP guidance.
117.	NRC Metrication Policy	Converted the cited value of 20°F to the metric equivalent of 11°C. Revised the text to cite the value in the dual unit format in accordance with the metrication policy and SRP-UDP guidance.
118.	NRC Metrication Policy	Converted the cited value of 75 ft-lbs to the metric equivalent of 102 J. Revised the text to cite the value in the dual unit format in accordance with the metrication policy and SRP-UDP guidance.
119.	Editorial	Added hyphens to "end of life."
120.	NRC Metrication Policy	Converted the cited value of 70 ft-lbs to the metric equivalent of 95 J. Revised the text to cite the value in the dual unit format in accordance with the metrication policy and SRP-UDP guidance.
121.	**Integrated Impact 446.**	Added a discussion of minimum fracture toughness requirements of 10 CFR 50, Appendix G, and the NUREG-0744 methods to assess the fracture toughness if the minimum requirements are not met.
122.	NRC Metrication Policy	Converted the cited value of 30°F to the metric equivalent of 17°C. Revised the text to cite the value in the dual unit format in accordance with the metrication policy and SRP-UDP guidance.
123.	NRC Metrication Policy	Converted the cited value of 50°F to the metric equivalent of 28°C. Revised the text to cite the value in the dual unit format in accordance with the metrication policy and SRP-UDP guidance.
124.	NRC Metrication Policy	Converted the cited value of 40°F to the metric equivalent of 22°C. Revised the text to cite the value in the dual unit format in accordance with the metrication policy and SRP-UDP guidance.
125.	Editorial	Corrected typographical error. The word "by" is changed to "be."
126.	**Integrated Impacts 447, 1132.**	Updated the reference to ASTM E-185 to reflect the latest version (1982) of the standard that is endorsed by the NRC in regulatory documents. The latest version of the standard is ASTM E-185-1994.

Item	Source	Description
127.	**Integrated Impact 445.**	The revision to Regulatory Guide 1.99 indicates that copper and nickel are of greater importance than phosphorous, and phosphorous is no longer considered in the chemistry factor as applied in determining the adjusted RT_{NDT} per the Regulatory Guide.
128.	Editorial	Revised text to reflect organizational changes in the NRC.

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SRP Draft Section 5.3.2

Attachment B - Cross Reference of Integrated Impacts

Integrated Impact No.	Issue	SRP Subsections Affected
445	Modify Acceptance Criteria and Review Procedures to reflect staff positions related to radiation embrittlement of reactor vessel materials.	II, and BTP MTEB 5-2
446	Revise Acceptance Criteria, Review Procedures and Branch Technical Position (BTP) MTEB 5-2 to incorporate 10 CFR 50.60 and NUREG-0744.	II, III, IV, VI, and BTP MTEB 5-2
447	Revise the BTP MTEB 5-2 reference to ASTM E-185 to specify the 1982 version.	BTP MTEB 5-2
658	Revise the SRP to cite the latest version of ASTM E208 (1991).	Placeholder - no changes to SRP.
972	Develop Acceptance Criteria, Review Procedures and Evaluation Findings for determining PWR reactor vessel acceptability under pressurized thermal shock (PTS) conditions.	I, II, III, IV, and VI
1123	Revise Acceptance Criteria and Review Procedures to incorporate the TMI action plan item II.K.2.13 related to analysis of thermal-mechanical conditions in the reactor vessel during recovery from small breaks with an extended loss of all feedwater.	Based on PRB comments,there are no changes to the SRP resulting from this integrated impact.
1131	Revise the Review Procedures to address review of plant specific temperature limits for combined operating license applicants.	III
1132	Revise the Branch Technical Position, BTP MTEB 5-2, to cite the latest version of ASTM E-185.	Placeholder - no changes to SRP.
1158	Revise the Acceptance Criteria, Review Procedures, and Evaluation Findings as necessary to incorporate the guidance of the proposed draft Regulatory Guide DG-1023.	Placeholder - no changes to SRP.
1160	Revise the Acceptance Criteria, Review Procedures, and Evaluation Findings as necessary to incorporate the guidance of the proposed draft Regulatory Guide DG-1025.	Placeholder - no changes to SRP.
1163	Revise the Acceptance Criteria, Review Procedures, and Evaluation Findings as necessary to incorporate the guidance of the proposed draft Regulatory Guide DG-1027.	Placeholder - no changes to SRP.
1206	Revise the Acceptance Criteria, Review Procedures and Evaluation Findings to incorporate the requirements from proposed rulemaking 59 FR 50513.	Placeholder - no changes to SRP.

1320	Revise Regulatory Guide 1.99 to address the issue related to the standard deviation used in calculating adjusted reference temperatures.	No changes to SRP. Further work to revise Regulatory Guide 1.99 is being tracked by IPD 7.0 form 5.3.2-2
1325	Revise Regulatory Guide 1.99 to address the issue related to the applicability of the Regulatory Guide to steels with high residual element concentrations.	No changes to SRP. Further work to revise Regulatory Guide 1.99 is being tracked by IPD 7.0 form 5.3.2-2
1378	Update the citation of ASTM E208 to cite the 1969 version.	VI.