U.S. NUCLEAR REGULATORY COMMISSION

STANDARD REVIEW PLAN

5.3.2 PRESSURE-TEMPERATURE LIMITS, UPPER-SHELF ENERGY, AND PRESSURIZED THERMAL SHOCK

REVIEW RESPONSIBILITIES

Primary - Organization responsible for the review of reactor vessel integrity issues related to the reactor coolant pressure boundary

Secondary - None

I. AREAS OF REVIEW

The staff will review the application with respect to the regulations concerning, (a) pressure-temperature (P-T) limits on maintaining the reactor coolant pressure boundary (RCPB), (b) reactor vessel beltline Charpy upper-shelf energy (USE), and (c) assessment of potential pressurized thermal shock (PTS) (pressurized-water reactor (PWR) only).

The specific areas of review are as follows:

1. Pressure-Temperature Limits/Upper-Shelf Energy/Pressurized Thermal Shock

   A. Pressure-Temperature Limits

The staff will review the P-T limits imposed on the RCPB during any condition of normal operation, including anticipated operational occurrences and hydrostatic tests, under this section of the SRP to ensure adequate safety margins of structural integrity for the ferritic components of the RCPB.

The regulations in 10 CFR 50.60 and associated Appendix G to 10 CFR Part 50 describe the conditions that require P-T limits and provide the general basis for these limits.

B. Upper-Shelf Energy

The staff will review reactor vessel beltline materials, which must have Charpy USE values of no less than 102 joules (J) (75 foot-pounds (ft-lb)) initially and must maintain USE values throughout the life of the vessel of no less than 68 J (50 ft-lb). If a material's USE values are projected to be less than 68 J, a safety analysis must be performed that will provide margins of safety against fracture equivalent to those described by Appendix G to Section XI of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (hereinafter referred to as the ASME Code). Reactor vessel beltline Charpy USE drop may be estimated using Regulatory Guide 1.99 when surveillance data are not available or are not applicable.

C. Pressurized Thermal Shock

The staff will evaluate PWR reactor vessel beltline materials to ensure adequate resistance to failure during PTS events. The staff will consider 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, any associated safety analyses performed to support reactor operation. Projected values of RT_{PTS} for PWR reactor vessel beltline materials are determined in accordance with 10 CFR 50.61.

2. Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC). For design certification (DC) and combined license (COL) reviews, the applicant's proposed information on the ITAAC associated with the systems, structures, and components (SSCs) related to this SRP section is reviewed in accordance with SRP Section 14.3, "Inspections, Tests, Analyses, and Acceptance Criteria - Design Certification." The staff recognizes that the review of ITAAC is performed after review of the rest of this portion of the application against acceptance criteria contained in this SRP section. Furthermore, the ITAAC are reviewed to assure that all SSCs in this area of review are identified and addressed as appropriate in accordance with SRP Section 14.3.

3. COL Action Items and Certification Requirements and Restrictions. COL action items may be identified in the NRC staff's final safety evaluation report (FSER) for each certified design to identify information that COL applicants must address in the application. Additionally, DCs contain requirements and restrictions (e.g., interface requirements) that COL applicants must address in the application. For COL applications referencing a DC, the review performed under this SRP section includes information provided in response to COL action items and certification requirements and restrictions pertaining to this SRP section, as identified in the FSER for the referenced certified design.

Review Interfaces

The listed SRP sections interface with this section as follows:

1. Review of the material characteristics of the RCPB and the reactor vessel, including fracture toughness properties and the material surveillance program, is performed under SRP Sections 5.2.3 and 5.3.1.

2. Review of the overpressure protection system for consistency with the P-T limits in Appendix G to 10 CFR Part 50 is performed under SRP Section 5.2.2.

3. Review of the peak reactor vessel wall fluence for the design life of the plant is performed under SRP Section 4.3.

The specific acceptance criteria and review procedures are contained in the referenced SRP sections.

II. ACCEPTANCE CRITERIA

Requirements

Acceptance criteria are based on meeting the relevant requirements of the following Commission regulations:

1. 10 CFR 50.55a, as it relates to quality standards for the design, fabrication, erection, and testing of SSCs important to safety.

2. 10 CFR 50.60, as it relates to compliance with the requirements of Appendix G to 10 CFR Part 50.

3. 10 CFR 50.61, as it relates to fracture toughness criteria for PWRs relevant to PTS events.

4. General Design Criterion (GDC) 1, found in Appendix A to 10 CFR Part 50, as it relates to quality standards for design, fabrication, erection, and testing.

5. GDC 14, as it relates to ensuring an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture of the RCPB.

6. GDC 31, as it relates to ensuring that the RCPB will behave in a nonbrittle manner and that the probability of rapidly propagating fracture is minimized.

7. GDC 32, as it relates to the reactor vessel materials surveillance program.

8. Appendix G to 10 CFR Part 50, as it relates to material testing and fracture toughness.

9. 10 CFR 52.47(b)(1), as it relates to ITAAC (for design certification) sufficient to assure that the SSCs in this area of review will operate in accordance with the certification.

10. 10 CFR 52.80(a)(1), as it relates to ITAAC (for combined licenses) sufficient to assure that the SSCs in this area of review have been constructed and will be operated in conformity with the license and the Commission's regulations.

SRP Acceptance Criteria

Specific SRP acceptance criteria acceptable to meet the relevant requirements of the NRC's regulations identified above are as follows for review described in Subsection I of this SRP section. The SRP is not a substitute for the NRC's regulations, and compliance with it is not required. However, an applicant is required to identify differences between the design features, analytical techniques, and procedural measures proposed for its facility and the SRP acceptance criteria and evaluate how the proposed alternatives to the SRP acceptance criteria provide acceptable methods of compliance with the NRC regulations.

1. Pressure-Temperature Limits

   A. Applicable Regulations, Codes, and Basis Documents

      The regulations in 10 CFR 50.60 and associated Appendix G to 10 CFR Part 50 describe the conditions that require P-T limits and provide the general basis for these limits. Appendix G specifically requires that P-T limits must be at least as conservative as limits obtained by following Appendix G to Section XI of the ASME Code during heatup, cooldown, and test conditions. Appendix G to 10 CFR Part 50 also requires additional safety margins when the reactor core is critical.

      Since the regulations may not have included specific fracture toughness testing requirements for the ferritic materials in the pressure-retaining components at the time some of the reactor facilities were designed and constructed, Branch Technical Position (BTP 5-2) (attached) describes procedures for making estimates and assumptions concerning the fracture toughness properties of materials in the older plants.

      Although Appendix G to Section III of the ASME Code is usually referenced with regard to facility design and construction, the reviewer should instead apply the provisions of Appendix G to Section XI of the ASME Code when using this SRP. The following provide the rationale for using Appendix G to Section XI of the ASME Code instead of Appendix G to Section III of the ASME Code:

      i. Appendix G to 10 CFR Part 50 specifically references Appendix G to Section XI to the ASME Code, and Appendix G to Section III to the ASME Code contains similar provisions.

      ii. The differences between Appendix G to Section XI of the ASME Code and Appendix G to Section III of the ASME Code have resulted from a series of ASME code cases, including N-588, N-640, and N-641. Appendix G to Section III of the ASME Code has not been updated since those code cases were developed. However, the staff expects that Appendix G of Section III of the ASME Code will be updated to be consistent with Appendix G to Section XI of the ASME Code.

B. Pressure-Temperature Requirements

Appendix G to 10 CFR Part 50 requires that the pressure-temperature (P-T) limits defined in that Appendix be at least as conservative as limits obtained by following the methods of analysis and the margins of safety of Appendix G of Section XI of the ASME Code, as stated below:

i. Pressure-Temperature Limits for Preservice Hydrostatic Tests

During preservice hydrostatic tests (if fuel is not in the vessel), a material's lower bound static crack initiation fracture toughness, K_{Ic}, must be greater than the K_I caused by pressure stresses acting on a defined, conservative hypothetical flaw, as shown in the following expression:

$$K_{applied} = K_I(\text{pressure}) < K_{Ic}$$

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

During performance of inservice leak and hydrostatic tests, a material's K_{Ic} must be greater than 1.5 times the K_I caused by pressure, as shown in the following expression:

$$K_{applied} = 1.5\ K_I(\text{pressure}) < K_{Ic}$$

iii. Pressure-Temperature Limits for Heatup and Cooldown Operations

At all times during heatup and cooldown operations, a material's K_{Ic} must be greater than the sum of 2 times the K_I caused by pressure and the K_I caused by thermal gradients, as shown in the following expression:

$$K_{applied} = 2K_I(\text{pressure}) + K_I(\text{thermal}) < K_{Ic}$$

iv. Pressure-Temperature Limits for Core Critical 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. In addition, the P-T relationship must provide at least a 22 °C (40 °F) margin over that required for heatup and cooldown operations.

2. Upper-Shelf Energy

A. Applicable Regulations, Codes, and Basis Documents

Appendix G to 10 CFR Part 50 requires that reactor vessel beltline materials have a Charpy USE value in the transverse direction for base material and along the weld for weld material according to the ASME Code of no less than 102 J (75 ft-lb) initially and must maintain a Charpy USE value throughout the life of the vessel of no less than 68 J (50 ft-lb), unless it is demonstrated in a manner approved by the Director, Office of Nuclear Reactor Regulation, that lower values of Charpy USE will provide margins of safety against fracture equivalent to those required by Appendix G of Section XI of the ASME Code.

B. Upper-Shelf Energy Requirements

Appendix G to 10 CFR Part 50 contains the following USE requirements:

i. Initially, the USE value in the transverse direction for base material and along the weld must not be less than 102 J (75 ft-lb).

ii. Charpy USE throughout the life of the vessel must be maintained at no less than 68 J (50 ft-lb), unless it is demonstrated in a manner approved by the Director, Office of Nuclear Reactor Regulation, that lower values of Charpy USE will provide margins of safety against fracture equivalent to those required by Appendix G of Section XI of the ASME Code.

3. Pressurized Thermal Shock

A. Applicable Regulations, Codes, and Basis Documents

Projected values of RT_{PTS} must be determined for PWR reactor vessel beltline materials in accordance with 10 CFR 50.61. 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 criterion is allowed.

B. Pressurized Thermal Shock Requirements

In accordance with 10 CFR 50.61, 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, throughout the facility's licensed operating permit. This assessment must be updated whenever projected values of RT_{PTS} change significantly, or upon request for a change in the expiration date for operation of the facility. 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 criterion is allowed.

Technical Rationale

The technical rationale for application of the above acceptance criteria to the areas of review addressed by this SRP section 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 SSCs important to safety. GDC 1 establishes that the quality standards to be applied to SSCs shall be commensurate with the importance of the safety functions to be performed. 10 CFR 50.55a establishes, in relevant part,

those provisions of the ASME 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 RCPB's primary safety functions include preventing a loss of reactor coolant through leakage or gross failure of RCPB piping or components, and acting as a containment barrier to the release of fission products in the event of an accident resulting in fuel damage. In accordance with Appendix G to Section III of the ASME Code, P-T limits are established for the RCPB to ensure the satisfaction of the RCPB material fracture toughness requirements. 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 that 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 must 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 the 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. The P-T 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 nonbrittle manner under operating, maintenance, testing, and postulated accident conditions. Application of GDC 14 to the RCPB, with regard to the P-T limits, provides assurance that the RCPB meets the material fracture toughness requirements and will act in a nonbrittle manner, thereby providing a low probability of significant degradation or of 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 must be designed with sufficient margin to ensure that when stressed under operating, maintenance, testing, and postulated accident conditions, the boundary behaves in a nonbrittle manner and the probability of rapidly propagating fracture is minimized. The design must 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 the size of flaws. The RCPB provides a fission product barrier, confinement of reactor coolant, and flowpaths 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 ensures that the P-T 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 the capability to contain reactor coolant inventory, reduction in the capability to confine fission products, and interference with core cooling is thereby minimized.

4. 10 CFR 50.60 requires that all light-water nuclear power reactors meet the fracture toughness requirements, including P-T limits, as set forth in Appendix G to 10 CFR Part 50. 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 next item discusses the technical rationale for this rule.

5. Appendix G to 10 CFR Part 50 establishes that the pressure-retaining components of the RCPB that are made of ferritic materials must meet requirements of the ASME Code, supplemented by the additional requirements set forth in Appendix G to 10 CFR Part 50 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 XI of the ASME Code, as supplemented by the requirements of Appendix G to 10 CFR Part 50. The P-T limits established in accordance with the ASME Code and Appendix G to 10 CFR Part 50 are used to establish operating parameters that provide assurance that the RCPB will act in a nonbrittle manner when subjected to stresses associated with normal operations, maintenance, testing, and anticipated operational occurrences. The P-T limits must be adjusted to account for the effects of radiation embrittlement of the RCPB materials over the life of the plant. Compliance with the requirements of Appendix G provides a method of satisfying the requirements of GDC 14 and 31 with regard to ensuring that the RCPB acts in a nonbrittle manner and that the probability of rapidly propagating failure and gross rupture of the RCPB is extremely low.

6. 10 CFR 50.61 establishes fracture toughness requirements for protection against PTS events, which involve transients in PWRs 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 material with low toughness. 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 for establishing the material reference temperature limits beyond which corrective actions and plant-specific safety analyses must justify continued operation of the plant. Establishing, monitoring, and maintaining the structural integrity of the reactor vessel materials are 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.

III. REVIEW PROCEDURES

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

For each area of review specified in Subsection I of this SRP section, the review procedure is identified below. These review procedures are based on the identified SRP acceptance criteria. For deviations from these specific acceptance criteria, the staff should review the applicant's evaluation of how the proposed alternatives to the SRP criteria provide an acceptable method of complying with the relevant NRC requirements identified in Subsection II.

1. Construction Permit/Design Certification Reviews

   The staff will review the information in the preliminary safety analysis report/standard safety analysis report for a commitment that the fracture toughness of the ferritic materials in the RCPB will comply with the requirements of Appendix G to

10 CFR Part 50, as detailed in Section XI of the ASME Code, and that the materials in the beltline region of the reactor vessel will comply with the requirements of 10 CFR 50.61 (PWRs only) and the guidance of Regulatory Guide 1.99.

2. Operating License/Combined License Reviews

The plant technical specifications or pressure-temperature limits report should show the P-T limits using real temperature. The staff will review these curves and their bases to determine acceptability in the following areas:

A. The limiting RT_{NDT} has been properly determined and radiation effects are included in an acceptable manner.

B. Limits are shown for all required conditions and provide all required information.

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

3. Acceptability Determination Methods

A. Pressure-Temperature Limits

The reviewer will perform an independent evaluation of one or more proposed P-T limit curves. The reviewer will base this evaluation on the methodology for constructing P-T limit curves found in Appendix G to Section XI of the most recent edition and addenda of the ASME Code that 10 CFR 50.55a has endorsed. The reviewer will also apply the additional minimum temperature requirements specified in Appendix G to 10 CFR Part 50.

For checking any P-T limit curve, the following steps describe the general form of the staff's evaluation.

i. Verify what each axis of the P-T limit plot represents. "Temperature," normally given on the horizontal axis, may be either the reactor coolant system fluid temperature (most common) or the metal temperature of the vessel. "Pressure," normally given on the vertical axis, is the system pressure but may be given in absolute or gauge values.

The reviewer should also check to see whether the curves include pressure and/or temperature measurement uncertainties have been included in the curves. If so, these must be removed before the evaluation outlined below will give comparable results.

ii. Determine the reference temperature (RT_{NDT}) at the 1/4 thickness (1/4 T) and 3/4 T locations for each vessel beltline material. For preservice hydrostatic testing curves, this determination shall be based on the initial material properties for each material determined in accordance with ASME Code, Section III, NB-2331 or BTP 5-2.

For all other curves (inservice leak/hydrostatic testing, heatup, cooldown, core critical operation), a period of applicability should be specified, usually in effective full-power years (EFPYs) of operation. This should be specified, along with the 1/4 T and 3/4 T neutron fluence (E > 1.0 MeV) at the EFPY value for each beltline material, to account for the effects of radiation on the material properties of each beltline material. The RT_{NDT} values at the 1/4 T and 3/4 T locations for each beltline material through the end of the specified period of applicability should be determined in accordance with Regulatory Guide 1.99.

iii. The following fundamental equation should be satisfied at each P-T point along any P-T limit curve:

$$K_{I\ applied} = K_{Ic}$$

where:

$K_{I\ applied}$ = The stress intensity due to pressure (membrane) and thermal gradient (bending) loads at the tip of the 1/4 T defect postulated in Appendix G to Section XI of the ASME Code.

K_{Ic} = The lower bound, plane strain, crack initiation fracture toughness for the material as represented in Figure 1.

iv. $K_{I\ applied}$ shall be calculated from an equation of the general form:

$$K_{I\ applied} = SF * M_m * (p * R_i / t) + K_{I\ thermal}$$

where:

SF = A structural factor applied to the pressure loading as specified in Appendix G to Section XI of the ASME Code and dependent on which P-T curve is being evaluated.

M_m = An influence coefficient to convert applied stress to crack tip stress intensity. M_m depends on the orientation of the flaw being evaluated (axial flaws for plates, forgings, and axial welds; circumferential flaws for circumferential welds) and the thickness of the material. Appendix G to Section XI of the ASME Code specifies values for M_m.

p = The pressure at the specified condition.

R_i = The vessel inside radius.

t = The vessel wall thickness.

$K_{I\ thermal}$ = The stress intensity at the crack tip due to thermal loadings (which are only considered for heatup, cooldown, and core critical operation curves). $K_{I\ thermal}$ may be conservatively calculated from equations given in Appendix G to Section XI of the ASME Code, which depend on heatup/cooldown rate and the vessel thickness. $K_{I\ thermal}$ may also be more accurately obtained from ORNL/NRC/LTR-03/03 for a given heatup/cooldown rate and the vessel thickness.

v. K_{Ic} is determined from Figure 1 (taken from Appendix G to Section XI of the ASME Code). K_{Ic} is a function of a material's RT_{NDT} value and temperature at the location of interest (i.e., either the 1/4 T or 3/4 T location).

It should be noted that a material's RT_{NDT} value will vary through the wall thickness as the neutron fluence decreases from the vessel inside diameter to the vessel outside diameter. The reviewer should apply the methods of Regulatory Guide 1.99 for determining the appropriate RT_{NDT} values.

It should also be noted that the temperature to be applied in using Figure 1 is the <u>metal</u> temperature at the tip of the postulated flaw from Appendix G to Section XI of the ASME Code (i.e., at either the 1/4 T or 3/4 T location). The metal temperature at throughwall locations will depend on the reactor coolant system fluid temperature and the rate of change of the reactor coolant system fluid temperature. Throughwall metal temperatures can be determined from methods given in Appendix G to Section XI of the ASME Code or from ORNL/NRC/LTR-03/03.

vi. Based on the discussion above, it is recommended that the following equation be solved to determine the allowable pressure associated with a specified temperature along a P-T limit curve:

$$\text{allowable pressure} = t * (K_{Ic} - K_{I\ thermal}) / (SF * M_m * R_i)$$

which is an algebraic rearrangement of the equation from (iv). The reviewer should keep in mind, however, that four of the quantities (K_{Ic}, $K_{I\ thermal}$, SF, and M_m) are dependent on other, more basic variables or conditions:

K_{Ic}—Depends on metal temperature and material RT_{NDT}

$K_{I\ thermal}$—Depends on heatup/cooldown rate and vessel wall thickness

SF—Depends on the curve being evaluated and the assumed flaw orientation

M_m—Depends on the vessel wall thickness and the assumed flaw orientation

vii. The reviewer should verify that all minimum temperature requirements specified in Appendix G to 10 CFR Part 50 for the P-T limit curve being verified have been met. These requirements have been imposed to ensure that highly stressed, nonbeltline regions (i.e., the vessel flange region) are protected from brittle failure.

viii. It should also be noted that some applications may provide P-T limit curves for other specific nonbeltline regions (e.g., the nozzle course and/or the bottom head of a boiling-water reactor) to address specific modes of operation. These nonbeltline curves are normally submitted to provide additional operational/testing flexibility and are generally less restrictive than the corresponding beltline curve. The development of these curves necessitates the evaluation of complex geometries where discontinuities are present. No simple review procedure can be specified for the review of such curves. However, the reviewer may use the guidance provided in Appendix G to Section XI of the ASME Code and Welding Research Council Bulletin 175 for such situations.

B. Upper-Shelf Energy

The reviewer will evaluate the initial Charpy USE values for the reactor vessel materials in accordance with the acceptance criterion specified in paragraph IV.A.1.a of Appendix G to 10 CFR Part 50. Reactor vessel materials that do not meet the specified initial Charpy USE acceptance criterion shall be evaluated in accordance with the provisions for additional analysis also specified in paragraph IV.A.1.a. In addition to the ASME Code, Regulatory Guide 1.161 provides an acceptable methodology for the performance of analyses intended to meet the provisions for the additional analysis in paragraph IV.A.1.a.

The reviewer will also evaluate the end-of-license (EOL) Charpy USE values for the reactor vessel materials in accordance with the acceptance criterion specified in paragraph IV.A.1.a of Appendix G to 10 CFR Part 50. Regulatory Guide 1.99 provides guidance on evaluating Charpy USE drop and, hence, USE values. Reactor vessel materials that do not meet the projected EOL Charpy USE minimum 68 J (50 ft-lb) criterion shall be evaluated in accordance with the provisions for additional analysis also specified in paragraph IV.A.1.a. In accordance with paragraph IV.A.1.c., this analysis must be submitted to the staff for review and approval on an individual case basis at least 3 years prior the date on which the predicted Charpy USE will no longer satisfy the requirements of paragraph IV.A.1.a, or on a schedule approved by the Director, Office of Nuclear Regulation. In addition to the ASME Code, Regulatory Guide 1.161 provides an acceptable methodology for the performance of analyses intended to meet the provisions for additional analysis specified in paragraph IV.A.1.a.

C. Pressurized Thermal Shock in Pressurized-Water Reactors

The reviewer will evaluate the projected values for RT_{PTS}, including the calculational methods and assumptions, and compare 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 should submit an analysis and schedule for the 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 expiration date of the operating license, the licensee can choose between the two options in 10 CFR 50.61 to meet PTS requirements. The licensee can submit a safety analysis to determine the modifications 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. The staff will review these safety analyses against the requirements of 10 CFR 50.61 and the guidance of Regulatory Guide 1.154. Alternatively, the licensee can perform a thermal-annealing treatment of the reactor vessel pursuant to 10 CFR 50.61(b)(7) to recover fracture toughness. In accordance with 10 CFR 50.61, the licensee must submit for NRC approval details of the approach selected at least 3 years before the reactor vessel is projected to exceed the PTS screening criteria.

4. For reviews of DC and COL applications under 10 CFR Part 52, the reviewer should follow the above procedures to verify that the design set forth in the safety analysis report, and if applicable, site interface requirements meet the acceptance criteria. For DC applications, the reviewer should identify necessary COL action items. With respect to COL applications, the scope of the review is dependent on whether the COL applicant references a DC, an ESP, or other NRC-approved material, applications, and/or reports.

   After this review, SRP Section 14.3 should be followed for the review of Tier I information for the design, including the postulated site parameters, interface criteria, and ITAAC.

IV. EVALUATION FINDINGS

The reviewer verifies that the applicant has provided sufficient information and that the review and calculations (if applicable) support conclusions of the following type to be included in the staff's safety evaluation report. The reviewer also states the bases for those conclusions.

> The pressure-temperature limits imposed on the reactor coolant system for all operating and testing conditions to ensure adequate safety margins against nonductile or rapidly propagating failure are in conformance with the fracture toughness criteria of Appendix G to 10 CFR Part 50 and Appendix G, "Protection Against Nonductile Failure," of the ASME Boiling and Pressure Vessel Code. The applicant has adequately addressed the upper-shelf energy criteria in accordance with Appendix G to 10 CFR Part 50 and thermal shock events in accordance with 10 CFR 50.61 [PWRs only]. The staff concludes that the use of operating limits, based upon the criteria defined in SRP Section 5.3.2, provides reasonable assurance that nonductile or rapidly propagating failure will not occur, and constitutes an acceptable basis for satisfying the applicable requirements of 10 CFR 50.55a, 10 CFR 50.60, and 10 CFR 50.61 [PWRs only] and GDC 1, 14, and 31 of Appendix A to 10 CFR Part 50.

For DC and COL reviews, the findings will also summarize (to the extent that the review is not discussed in other SER sections) the staff's evaluation of the ITAAC, including design acceptance criteria, as applicable, and interface requirements and combined license action items relevant to this SRP section.

V. IMPLEMENTATION

The staff will use this SRP section in performing safety evaluations of DC applications and license applications submitted by applicants pursuant to 10 CFR Part 50 or 10 CFR Part 52. Except when the applicant proposes an acceptable alternative method for complying with specified portions of the Commission's regulations, the staff will use the method described herein to evaluate 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, unless superceded by a later revision.

VI. REFERENCES

1. 10 CFR 52.47, "Contents of Applications."

2. 10 CFR 50.55a, "Codes and Standards."

3. 10 CFR 50.60, "Acceptance Criteria for Fracture Prevention Measures for Lightwater Nuclear Power Reactors for Normal Operation."

4. 10 CFR 50.61, "Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events."

5. 10 CFR Part 50, Appendix A, General Design Criterion 1, "Quality Standards and Records."

6. 10 CFR Part 50, Appendix A, General Design Criterion 14, "Reactor Coolant Pressure Boundary."

7. 10 CFR Part 50, Appendix A, General Design Criterion 31, "Fracture Prevention of Reactor Coolant Pressure Boundary."

8. 10 CFR Part 50, Appendix A, General Design Criterion 32, "Inspection of Reactor Coolant Pressure Boundary."

9. 10 CFR Part 50, Appendix G, "Fracture Toughness Requirements."

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

11. 10 CFR Part 52, "Early Site Permits; Standard Design Certifications; and Combined Licenses for Nuclear Power Plants."

12. 10 CFR 52.97, "Issuance of Combined Licenses."

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

14. ASME Boiler and Pressure Vessel Code, Section XI, Division 1, Appendix G, " Fracture Toughness Criteria for Protection Against Failure," American Society of Mechanical Engineers.

15. Branch Technical Position BTP 5-2, “Fracture Toughness Requirements,” attached to this SRP subsection.

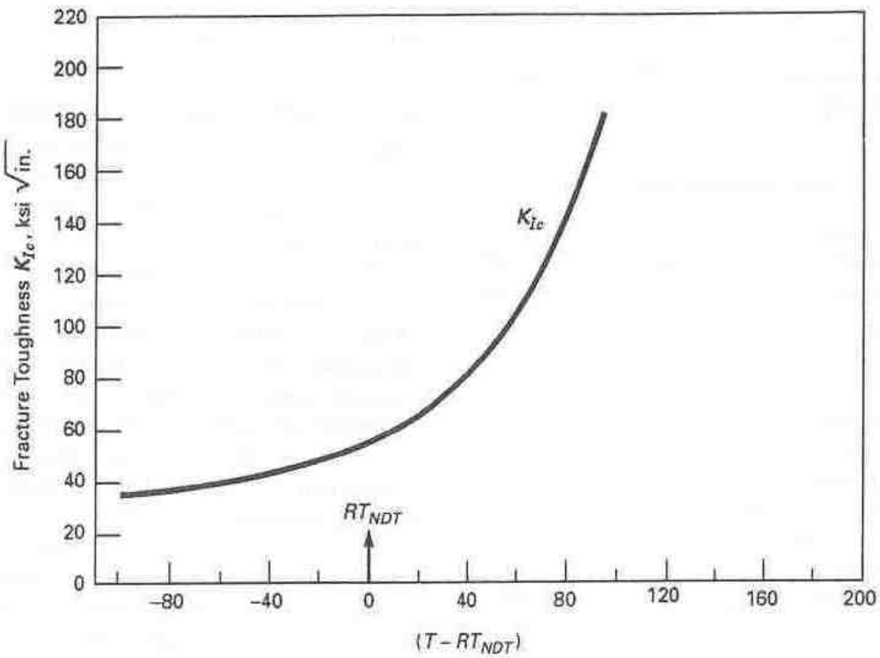
16. ORNL/NRC/LTR-03/03, “Tabulation of Thermally-Induced Stress Intensity Factors (K_{IT}) and Crack Tip Temperatures for Generating P-T Curves per ASME Section XI—Appendix G.”

17. Regulatory Guide 1.99, “Radiation Embrittlement of Reactor Vessel Materials.”

18. Regulatory Guide 1.154, “Format and Content of Plant-Specific Pressurized Thermal Shock Safety Analysis Reports for Pressurized Water Reactors.”

19. Regulatory Guide 1.161, “Evaluation of Reactor Pressure Vessels with Charpy Upper-Shelf Energy Less Than 50 Ft-Lb.”

20. Welding Research Council Bulletin 175, “PVRC Recommendation on Fracture Toughness,” Welding Research Council, Pressure Vessel Research Committee Ad Hoc Group on Toughness Requirements, August 1972.



K_{Ic}=33.2+20.734 exp [0.02(T-RT_{NDT})]

Temperature Relative to RT_{NDT}, (T-RT_{NDT}), Fahrenheit Degrees

FIGURE 1

BRANCH TECHNICAL POSITION BTP 5-2
FRACTURE TOUGHNESS REQUIREMENTS

A. Background

NRC requirements regarding fracture toughness, pressure-temperature limits, material surveillance, and pressurized thermal shock (PTS) (PWR only) are contained in Appendices A, G, and H to 10 CFR Part 50 and in 10 CFR 50.61; these requirements also refer to relevant sections of the ASME Code. The purpose of this branch technical position is to summarize these requirements and provide guidance, 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.

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 the criteria of 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 should be assessed by using the available test data to estimate the fracture toughness in the same terms as the new requirements. This should be done because the operating limitations imposed on old plants should 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 results of fracture toughness tests. Both drop weight nil-ductility transition temperature (NDTT) tests and Charpy V-notch tests should be run to determine the RT_{NDT}. The NDTT temperature, as determined by drop weight tests (ASTM E-208-1969) is the RT_{NDT} if, at 33 °C (60 °F) above the NDTT, at least 68 J (50 ft-lbs) of energy and 0.89 mm (35 mils) lateral expansion (LE) are obtained in Charpy V-notch tests on specimens oriented in the weak direction (transverse to the direction of maximum working).

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

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

(2) If dropweight tests were not performed on SA-508, Class II forgings, the NDTT may be estimated as the lowest of the following temperatures:

(a) 33 °C (60 °F).

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

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

(3) If transversely-oriented Charpy V-notch specimens were not tested, the temperature at which 68 J (50 ft-lbs) and 0.89 mm (35 mils) LE would have been obtained on transverse specimens may be estimated by one of the following criteria:

(a) Test results from longitudinally-oriented specimens reduced to 65% of their value to provide conservative estimates of values expected from transversely oriented specimens.

(b) Temperatures at which 68 J (50 ft-lbs) and 0.89 mm (35 mils) LE were obtained on longitudinally-oriented specimens increased 11 °C (20 °F) to provide a conservative estimate of the temperature that would have been necessary to obtain the same values on transversely-oriented specimens.

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

1.2 Estimation of Charpy V-Notch Upper Shelf Energies

For the beltline region of reactor vessels, the upper shelf toughness must account for the effects of neutron radiation. Reactor vessel beltline materials must have Charpy upper shelf energy, in the transverse direction for base material and along the weld for weld material according to the ASME Code, of no less than 102 J (75 ft-lbs) initially and must maintain Charpy upper shelf energy throughout the life of the vessel of no less than 68 J (50 ft-lbs).

If Charpy upper shelf 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 Part 50, Appendix G, paragraph V.C.3.

1.3 Reporting Requirements

Fracture toughness information identified by the Code and by Appendix G, 10 CFR Part 50, should 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 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) For system and component hydrostatic tests performed prior to loading fuel in the reactor vessel, it is recommended that hydrostatic tests be performed at a temperature not lower than RT_{NDT} plus 60 °F.

(3) For system and component hydrostatic tests performed subsequent to loading fuel in the reactor vessel, the minimum test temperature should be determined as discussed in Section 3 of SRP 5.3.2.

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 the report, "Tabulation of thermally-Induced Stress Intensity Factors (K_{IT}) and Crack Tip Temperatures for Generating P-T Curves per ASME Section XI-Appendix G," ORNL/NRC/LTR-03/03.

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) 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 should 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 over conservative assumptions and procedures.

2.3 Reporting Requirements

The Technical Specifications should include the operating and test limits discussed above, and the basis for their determination. The Technical Specifications should 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. The selection of material to be included in the surveillance program should be in accordance with ASTM E-185-82, 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 nickel content, for example) and the neutron fluence expected at its location in the vessel.

3.2 SAR Criteria

With respect to the adequacy of the surveillance program, information requested for beltline materials includes the following:

(1) Tensile properties.

(2) Dropweight test 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 nickel 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 should be included in the Technical Specifications.

3.4 Reporting Criteria

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 Office of Nuclear Reactor Regulation for approval.

4. Pressurized Thermal Shock (PWR only)

4.1 Pressurized Thermal Shock Requirements

As required by 10 CFR 50.61, the following is a summary of requirements for the PWR reactor vessels:

(1) RT_{PTS} values must be projected using end-of-life fluence for each weld, plate or forging in the reactor vessel beltline region. The projected EOL RT_{PTS} values must be approved by the NRC.

(2) PTS screening criteria is 132 °C (270 °F) for plates, forgings, and axial weld materials, and 149 °C (300 °F) for circumferential weld materials.

(3) If reactor vessel is projected to exceed the PTS screening criteria, 10 CFR 50.61(b)(3) requires the applicant to implement a flux reduction

program that is reasonably practicable to avoid exceeding the PTS screening criteria.

(4) If the flux reduction program does not prevent the reactor vessel from exceeding the PTS screening criterion at the end of life, the applicant choose between the two options in 10 CFR 50.61 to meet PTS requirements: (a) submit a safety analysis pursuant to 10 CFR 50.61(b)(4) to determine what, if, any, modifications to equipment, systems, and plant operation to prevent failure of the reactor vessel from a postulated PTS event, (b) perform a thermal-annealing treatment of the reactor vessel pursuant 10 CFR 50.61(b)(7) to recover fracture toughness. 10 CFR 50.61 requires details of the approach selected to be submitted for NRC approval at least 3 years before the reactor vessel is projected to exceed the PTS screening criteria.

SRP Section 5.3.2
Description of Changes

This SRP section affirms the technical accuracy and adequacy of the guidance previously provided in Draft Revision 2, dated June 1996, of this SRP. See ADAMS accession number ML406408232.

In addition, this SRP section was administratively updated in accordance with NRR Office Instruction, LIC-200, Revision 1, “Standard Review Plan (SRP) Process.” The revision also adds standard paragraphs to extend application of the updated SRP section to prospective submittals by applicants pursuant to 10 CFR Part 52.

The technical changes are incorporated in Revision 2, dated [Month] 2007.

Review Responsibilities - Reflects changes in review branches resulting from reorganization and branch consolidation. Change is reflected throughout the SRP.

Replaced specific organization title with organizational function to allow future reorganization and branch consolidation without necessitating an SRP change. Change is consistent with updated SRP guidance and is reflected throughout the SRP.

I. AREAS OF REVIEW

1. Reorganized this section to differentiate requirements for P-T limits, USE, and PTS.

2. Added review requirements for reactor vessel Charpy USE in Subsection I.B.

II. ACCEPTANCE CRITERIA

1. Added Appendix G to 10 CFR Part 50 for USE criteria.

2. Reorganized this section to include discussions on P-T limits, USE, and PTS.

3. Provided explanation of using Appendix G to Section XI of the ASME Code instead of Appendix G to Section III. Appendix G to Section XI has been updated to reflect the code cases and is currently used for the development of P-T limits. It is expected that Appendix G to Section III of the ASME Code will also be updated to reflect the criterion in Appendix G to Section XI.

4. Used K_{Ic} in the development of P-T limits in accordance with the latest ASME Code instead of K_{Ia}.

5. Added discussions on USE to provide (1) applicable regulations, codes, and basis documents and (2) USE requirements.

6. Revised the subheading to accommodate the 10 CFR Part 52 licensing process with regard to the review of P-T limits.

III. REVIEW PROCEDURES

1. Revised the acceptability determination methods for P-T limits to provide a detailed step-by-step process to review the acceptance of P-T limits. Resulted in no change in methodology except the use of K_{Ic} instead of K_{Ia}.

2. Added a paragraph to address review criteria for the reactor vessel beltline Charpy USE criteria.

3. Added a paragraph to address the performance of DC and COL reviews pursuant to 10 CFR Part 52.

IV. EVALUATION FINDINGS

Added the last paragraph to address the performance of DC and COL reviews pursuant to 10 CFR Part 52.

V. IMPLEMENTATION

Added boilerplate text to implementation subsection to incorporate 10 CFR Part 52 and to address relevance of the section to existing and future applications.

VI. REFERENCES

Updated references to reflect applicable regulations and guidance and renumbered in accordance with the updated SRP format.