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 component integrity issues related to reactor vessels

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.

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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 staff reviews the applicant's proposed ITAAC associated with the structures, systems, and components (SSCs) related to this SRP section in accordance with SRP Section 14.3, "Inspections, Tests, Analyses, and Acceptance Criteria." The staff recognizes that the review of ITAAC cannot be completed until after the rest of this portion of the application has been reviewed against acceptance criteria contained in this SRP section. Furthermore, the staff reviews the ITAAC to ensure 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. For a DC application, the review will also address COL action items and requirements and restrictions (e.g., interface requirements and site parameters).

   For a COL application referencing a DC, a COL applicant must address COL action items (referred to as COL license information in certain DCs) included in the referenced DC. Additionally, a COL applicant must address requirements and restrictions (e.g., interface requirements and site parameters) included in the referenced DC.

Review Interfaces

Other 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), which requires that a DC application contain the proposed inspections, tests, analyses, and acceptance criteria (ITAAC) that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the design certification is built and will operate in accordance with the design certification, the provisions of the Atomic Energy Act, and the NRC's regulations;

10. 10 CFR 52.80(a), which requires that a COL application contain the proposed inspections, tests, and analyses, including those applicable to emergency planning, that the licensee shall perform, and the acceptance criteria that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are

performed and the acceptance criteria met, the facility has been constructed and will operate in conformity with the combined license, the provisions of the Atomic Energy Act, and the NRC'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 the review described in 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 5-3) 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 these 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 material from the procedures described below, as may be appropriate for a particular case.

These review procedures are based on the identified SRP acceptance criteria. For deviations from these acceptance criteria, the staff should review the applicant's evaluation of how the proposed alternatives 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 5-3.

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.

For review of a DC application, the reviewer should follow the above procedures to verify that the design, including requirements and restrictions (e.g., interface requirements and site parameters), set forth in the final safety analysis report (FSAR) meets the acceptance criteria. DCs have referred to the FSAR as the design control document (DCD). The reviewer should also consider the appropriateness of identified COL action items. The reviewer may identify additional COL action items; however, to ensure these COL action items are addressed during a COL application, they should be added to the DC FSAR.

For review of a COL application, the scope of the review is dependent on whether the COL applicant references a DC, an early site permit (ESP) or other NRC approvals (e.g., manufacturing license, site suitability report or topical report).

For review of both DC and COL applications, SRP Section 14.3 should be followed for the review of ITAAC. The review of ITAAC cannot be completed until after the completion of this section.

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 the staff's evaluation of requirements and restrictions (e.g., interface requirements and site parameters) and COL action items relevant to this SRP section.

In addition, to the extent that the review is not discussed in other SER sections, the findings will summarize the staff's evaluation of the ITAAC, including design acceptance criteria, as applicable.

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 submitted six months or more after the date of issuance of this SRP section, unless superseded 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 5-3, “Fracture Toughness Requirements”.

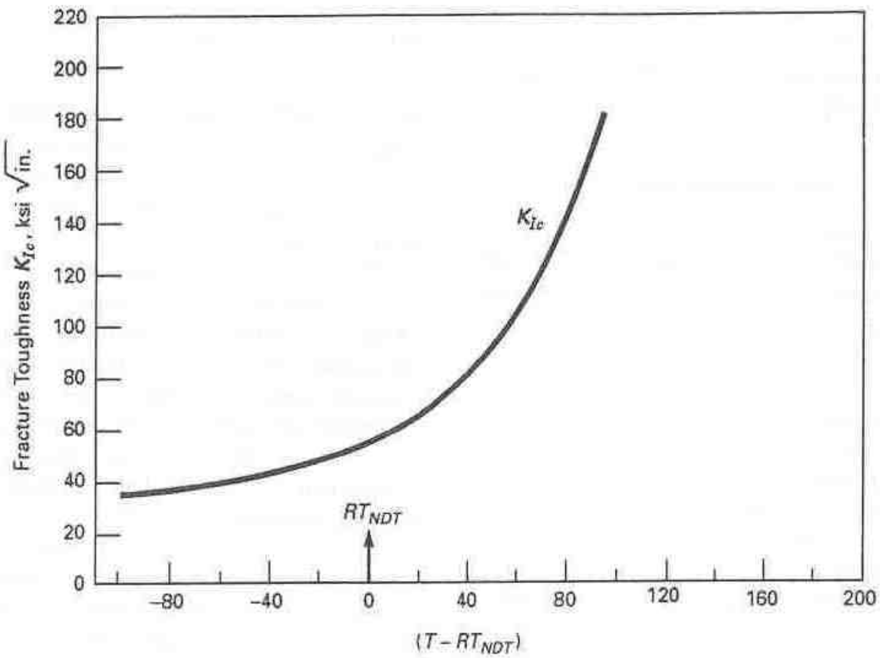
16. ORNL/NRC/LTR-03/03, “Tabulation of Thermally-Induced Stress Intensity Factors (K_{IT}) and Crack Tip Temperatures for Generating P-T Curves per ASME Section XI - Appendix G.”

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