NUREG-0800

U.S. NUCLEAR REGULATORY COMMISSION

STANDARD REVIEW PLAN

BRANCH TECHNICAL POSITION 5-3

FRACTURE TOUGHNESS REQUIREMENTS

REVIEW RESPONSIBILITIES

Primary - Organization responsible for the review of component integrity issues related to reactor vessels

Secondary - None

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.

Revision 2 - March 2007

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 III 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.

C. REFERENCES

1. 10 CFR Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants."

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

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

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

5. NUREG-0744, "Pressure Vessel Material Fracture Toughness."

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

7. ASTM E-185-82, "Standard Practice for Design of Surveillance Programs for Light-Water Moderated Nuclear Power Reactor Vessels."

8. 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."