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

The U.S. Nuclear Regulatory Commission (NRC) requirements regarding fracture toughness, pressure-temperature limits, material surveillance, and pressurized thermal shock (PTS) (pressurized-water reactor (PWR) only) are contained in Appendices A, G, and H to Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, "Domestic Licensing of Production and Utilization Facilities"; and in 10 CFR 50.61 "Fracture toughness requirements for protection against pressurized thermal shock events," or 10 CFR 50.61a, "Alternate fracture toughness requirements for protection against pressurized thermal shock events." These requirements also refer to relevant sections of the American Society of Mechanical Engineers (ASME) Code as incorporated by reference in 10 CFR 50.55a, "Codes and standards" (hereafter the ASME Code). Some sites have implemented Pressure Temperature Limit Reports (PTLR), approved via license amendment consistent with the guidance in Generic Letter (GL) 96-03, "Relocation of the Pressure Temperature Limit Curves and Low Temperature Overpressure Protections System Limits," which relocate portions of the subject requirements from the Technical Specifications (TS). 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 the requirements to these 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.

In a memo dated January 30, 2014, industry found that one of the methodologies for conservatively estimating reactor vessel material properties provided in B1.1 of this branch technical position was potentially non-conservative. In response, both industry and the NRC evaluated the methodologies provided under B1.1 and B1.2 for estimating material properties and assessed the impact of potential non-conservatisms on operating plants regarding pressure-temperature limits and PTS. The NRC's assessment was documented in an April 11, 2017 memorandum with attachments assessing the industry's two summary reports on this issue and addressing all technical issues related to B1.1 and B1.2. Based on the change-in-risk analyses documented in the memorandum, the NRC determined that the methodologies in B1.1 and B1.2 can be used in pressure-temperature limits and PTS evaluations for up to 72 effective full power years of operation.

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, "Fracture Toughness Criteria for Protection Against Failure," 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 approach ensures that the operating limitations imposed on old plants 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}, which 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) 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 the manner described above. For these older plants, 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 drop weight 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 drop weight 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 are reduced to 65 percent 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 are 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 in accordance with 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). Beltline materials are defined to be those materials directly surrounding the effective height of the active core and adjacent materials estimated to receive a neutron fluence of 1×10^{17} n/cm^2 (E > 1.0 MeV) or higher. This definition, which is derived from the definition of beltline in 10 CFR 50.61 and considers the neutron fluence zone of interest as defined in Appendix H(III)(a), 10 CFR Part 50, is consistent with NRC

Regulatory Issue Summary 2014-11, "Information on Licensing Applications for Fracture Toughness Requirements for Ferritic Reactor Coolant Pressure Boundary Components."

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 percent of the longitudinal values to estimate the transverse properties.

The predicted end-of-license (EOL) Charpy upper shelf energy and adjusted reference temperature for the reactor vessel materials must meet the requirements of 10 CFR Part 50, Appendix G, paragraph IV.A. Reactor vessel materials that do not meet the specified EOL acceptance criteria should be reviewed through an equivalent margins analysis in accordance with the same paragraph of 10 CFR Part 50, Appendix G. This equivalent margins analysis should be consistent with the guidance in Regulatory Guide 1.161, "Evaluation of Reactor Pressure Vessels with Charpy Upper-Shelf Energy Less Than 50 Ft-Lb," which references ASME Code, Section XI, Appendix K, "Assessment of Reactor Vessels with Low Upper Shelf Charpy Impact Energy Levels."

1.3 Reporting Requirements

Fracture toughness information identified by the ASME Code and by Appendix G, 10 CFR Part 50, should be reported in the final safety analysis report (FSAR) to provide a basis for evaluating the adequacy of the operating limitations given in the TS or PTLR. 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 TS or PTLR. The basis for the determination shall be reported by the applicant. 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 XI of the ASME 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 specified in Appendix G, 10 CFR Part 50.

2.2.2 Heatup and Cooldown Limit Curves

Heatup and cooldown pressure-temperature limit curves may be determined using the method given in Appendix G of the ASME Code. The effect of thermal gradients may be conservatively approximated by the procedures in Appendix G of the ASME 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 and if the nozzles are not limiting.

Alternatively, more rigorous analytical procedures may be used, provided that the intent of the ASME 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 TS.

The safety analysis report (SAR) should present descriptions of the continued integrity of all vital components of the Reactor Coolant Pressure Boundary (RCPB) during postulated faulted conditions. These descriptions should be made in as realistic a manner as possible, avoiding grossly over conservative assumptions and procedures.

2.3 Reporting Requirements

The TS should include the operating and test limits discussed above, and the basis for their determination. The TS 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. Sites with an approved PTLR must remain consistent with the requirements delineated on this topic concerning both the TS and PTLR.

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, “Reactor Vessel Material Surveillance Program Requirements,” 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, as incorporated by reference into Appendix H of 10 CFR Part 50, 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.

Some plants have received approval to join Integrated Surveillance Programs (ISP) in which case their surveillance program requirements are delineated in the relevant NRC safety evaluation for ISP approvals.

3.2 Safety Analysis Report 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, and/or consistent with an approved ISP. The proposed removal and test schedule should be included in the TS or PTLR.

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 NRC Office of Nuclear Reactor Regulation (NRR) for approval per 10 CFR 50 Appendix G. Sites with an approved PTLR may update the PTLR without NRC approval provided they do not deviate from the PTLR methodology approved by the NRC when the PTLR was implemented. Sites with an approved methodology must, when changes are made to the figures, values, and parameters contained in the PTLR, update and submit the PTLR to the NRC for information.

4. Pressurized Thermal Shock (PWR only)

4.1 Pressurized Thermal Shock Requirements

The following is a summary of requirements in 10 CFR 50.61 for the PWR reactor vessels:

(1) RT_{PTS} values must be projected using EOL 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 EOL, the applicant must 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 are necessary 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.

(5) Since 2010, another option became available in 10 CFR 50.61a to use the alternative PTS screening criteria for plants which could not prevent their reactor vessels from exceeding the PTS screen criteria of 10 CFR 50.61 at the EOL.

4.2 Alternative Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events

For PWRs for which a construction permit was issued under 10 CFR Part 50 before February 3, 2010, and whose reactor vessel was designed and fabricated to the 1998 Edition or earlier of the ASME Code, the requirements of 10 CFR 50.61a may be applied in lieu of 10 CFR 50.61. To apply 10 CFR 50.61a, the license holder must apply for and receive approval, via license amendment, from the NRR Director. The following is a summary of the requirements in 10 CFR 50.61a:

(1) Each licensee is to project values of RT_{MAX-X} for each reactor vessel beltline material for the EOL fluence of the material, where the subscript “X” could be axial weld (AX), plate (PL), forging (FO), or circumferential weld (CW).

(2) Each licensee shall perform an examination and assessment of flaws in the reactor vessel beltline as required by 10 CFR 50.61a(e).

(3) Each licensee shall compare the projected RT_{MAX-X} values for plates, forgings, axial welds, and circumferential welds to the PTS screening criteria of Table 1 of 10 CFR 50.61a.

C. REFERENCES

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

2. American Society of Mechanical Engineers, *ASME Boiler and Pressure Vessel Code*, Section XI, Appendix K, "Assessment of Reactor Vessels with Low Upper Shelf Charpy Impact Energy Levels."

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

4. American Society for Testing and Materials, ASTM E-185-82, "Standard Practice for Design of Surveillance Programs for Light-Water Moderated Nuclear Power Reactor Vessels."

5. U.S. *Code of Federal Regulations*, "Domestic Licensing of Production and Utilization Facilities," Part 50, Title 10, "Energy," Appendix A, "General Design Criteria for Nuclear Power Plants."

6. U.S. *Code of Federal Regulations*, "Domestic Licensing of Production and Utilization Facilities," Part 50, Title 10, "Energy," Appendix G, "Fracture Toughness Requirements."

7. U.S. *Code of Federal Regulation*s, "Domestic Licensing of Production and Utilization Facilities," Part 50, Title 10, "Energy," Appendix H, "Reactor Vessel Material Surveillance Program Requirements."

8. *U.S. Code of Federal Regulations,* "Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events," § 50.61, Title 10, "Energy."

9. *U.S. Code of Federal Regulations,* Alternative Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events. § 50.61a, Title 10, "Energy."

10. U.S. Nuclear Regulatory Commission, Generic Letter 96-03, "Relocation of the Pressure Temperature Limit Curves and Low Temperature Overpressure Protections System Limits."

11. U.S. Nuclear Regulatory Commission/Oak Ridge National Laboratory, "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, March 2003, (ADAMS Accession No. ML110070355).

12. U.S. Nuclear Regulatory Commission, Regulatory Guide 1.161, "Evaluation of Reactor Pressure Vessels with Charpy Upper-Shelf Energy Less Than 50 Ft-Lb."

13. Salas, Pedro, Areva Memorandum to U.S. Regulatory Commission, "Potential Non-Conservatism in NRC Branch Technical Position 5-3," January 30, 2014 (ADAMS Accession No. ML14038A265).

14. Rudland, David Memorandum to John Lubinski, Joseph Giitter, and George Wilson, U.S. Nuclear

Regulatory Commission, Closure Memorandum Supporting the Limited Revision of NUREG-0800 Branch Technical Position 5-3, “Fracture Toughness Requirements,” April 11, 2017 (ADAMS Accession No. ML16364A285).

Paperwork Reduction Act

This Standard Review Plan provides guidance for implementing the mandatory information collections in 10 CFR Parts 50 and 52 that are subject to the Paperwork Reduction Act of 1995 (44 U.S.C. 3501 et. seq.). These information collections were approved by the Office of Management and Budget, approval numbers 3150-0011 and 3150-0151. Send comments regarding this information collection to the Information Services Branch (T-2 F43), U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, or by e-mail to Infocollects.Resource@nrc.gov, and to the OMB reviewer at: OMB Office of Information and Regulatory Affairs (3150-0011 and 3150-0151), Attn: Desk Officer for the Nuclear Regulatory Commission, 725 17th Street, NW Washington, DC 20503; e-mail: oira_submission@omb.eop.gov.

Public Protection Notification

The NRC may not conduct or sponsor, and a person is not required to respond to, a collection of information unless the document requesting or requiring the collection displays a currently valid OMB control number.

BTP 5-3, “Fracture Toughness Requirements”
Description of Changes

The technical changes incorporated in BTP 5-3, “Fracture Toughness Requirements,” Revision 3, dated July 2018 include incorporation of text describing NRC review of potential non-conservatisms in BTP 5-3, Revision 2, Subsection B1.1; a reference to a memorandum describing the results of the NRC review of the potential non-conservatisms; and numerous textual updates to incorporate pressure-temperature limit reports, 10 CFR 50.61a, 10 CFR Part 52, “Licenses, Certifications, and Approvals for Nuclear Power Plants,” clearer citations, and nozzle language.

Descriptions of the changes in each BTP section are as follows:

A. BACKGROUND

1. Added discussion of Pressure Temperature Limit Reports (PTLR) and associated citation of GL 96-03

2. Added paragraph describing review of potential non-conservatisms

B. BRANCH TECHNICAL POSITION

1. Added clarifying NRC definition of “beltline materials” (B1.2)

2. Added reference to PTLR (B1.3, B2.1, B2.3)

3. Updated outdated citation of Section III of ASME Code to Section XI (B2.1)

4. Added note concerning potential for limiting nozzles (B2.2.2)

5. Added clarification concerning sites with approved Integrated Surveillance Programs (B3.1, B3.3)

6. Added clarification regarding updating operating and test limits for sites with approved PTLR (B3.5)

7. Added discussion of 10 CFR 50.61a as alternative Pressure Temperature Shock (PTS) screening criteria (B4.1(5))

8. Added discussion of Alternative Fracture Toughness Requirements for protection against PTS (B4.2)

C. REFERENCES

1. Added reference to NRC Memorandum (ADAMS Accession No. ML16364A285) concerning review of potential non-conservatisms.

2. Added reference to GL 96-03

3. Added reference to January 30, 2014 industry memo concerning non-conservatism of BTP 5-3