# §§50.61 Fracture toughness requirements for protection against pressurized thermal shock events.

(a) *Definitions.* For the purposes of this section:

(1) *ASME Code* means the American Society of Mechanical Engineers Boiler and Pressure Vessel Code, Section III, Division I, "Rules for the Construction of Nuclear Power Plant Components," edition and addenda and any limitations and modifications thereof as specified in §§50.55a.

(2) *Pressurized Thermal Shock Event* means an event or transient in pressurized water reactors (PWRs) causing severe overcooling (thermal shock) concurrent with or followed by significant pressure in the reactor vessel.

(3) *Reactor Vessel Beltline* means the region of the reactor vessel (shell material including welds, heat affected zones and plates or forgings) that directly surrounds the effective height of the active core and adjacent regions of the reactor vessel that are predicted to experience sufficient neutron radiation damage to be considered in the selection of the most limiting material with regard to radiation damage.

(4) *$RT_{NDT}$* means the reference temperature for a reactor vessel material, under any conditions. For the reactor vessel beltline materials, $RT_{NDT}$ must account for the effects of neutron radiation.

(5) *$RT_{NDT(U)}$* means the reference temperature for a reactor vessel material in the pre-service or unirradiated condition, evaluated according to the procedures in the ASME Code, Paragraph NB - 2331 or other methods approved by the Director, Office of Nuclear Reactor Regulation.

(6) *EOL Fluence* means the best-estimate neutron fluence projected for a specific vessel beltline material at the clad-base-metal interface on the inside surface of the vessel at the location where the material receives the highest fluence on the expiration date of the operating license.

(7) *$RT_{PTS}$* means the reference temperature, $RT_{NDT}$, evaluated for the EOL Fluence for each of the vessel beltline materials, using the procedures of paragraph (c) of this section.

(8) *PTS Screening Criterion* means the value of $RT_{PTS}$ for the vessel beltline material above which the plant cannot continue to operate without justification.

(b) *Requirements.* (1) For each pressurized water nuclear power reactor for which an operating license has been issued, other than a nuclear power reactor facility for which the certifications required under §§50.82(a)(1) have been submitted, the licensee shall have projected values of $RT_{PTS}$, accepted by the NRC, for each reactor vessel beltline material for the EOL fluence of the material. The assessment of $RT_{PTS}$ must use the calculation procedures given in paragraph (c)(1) of this section, except as provided in paragraphs (c)(2) and (c)(3) of this section. The assessment must specify the bases for the projected value of $RT_{PTS}$ for each vessel beltline material, including the assumptions regarding core loading patterns, and must specify the copper and

nickel contents and the fluence value used in the calculation for each beltline material. This assessment must be updated whenever there is a significant(2) change in projected values of $RT_{PTS}$, or upon request for a change in the expiration date for operation of the facility.

2) The pressurized thermal shock (PTS) screening criterion is 270 F for plates, forgings, and axial weld materials, and 300 F for circumferential weld materials. For the purpose of comparison with this criterion, the value of $RT_{PTS}$ for the reactor vessel must be evaluated according to the procedures of paragraph (c) of this section, for each weld and plate, or forging, in the reactor vessel beltline. $RT_{PTS}$ must be determined for each vessel beltline material using the EOL fluence for that material.

(3) For each pressurized water nuclear power reactor for which the value of $RT_{PTS}$ for any material in the beltline is projected to exceed the PTS screening criterion using the EOL fluence, the licensee shall implement those flux reduction programs that are reasonably practicable to avoid exceeding the PTS screening criterion set forth in paragraph (b)(2) of this section. The schedule for implementation of flux reduction measures may take into account the schedule for submittal and anticipated approval by the Director, Office of Nuclear Reactor Regulation, of detailed plant-specific analyses, submitted to demonstrate acceptable risk with $RT_{PTS}$ above the screening limit due to plant modifications, new information or new analysis techniques.

(4) For each pressurized water nuclear power reactor for which the analysis required by paragraph (b)(3) of this section indicates that no reasonably practicable flux reduction program will prevent $RT_{PTS}$ from exceeding the PTS screening criterion using the EOL fluence, the licensee shall submit a safety analysis to determine what, if any, modifications to equipment, systems, and operation are necessary to prevent potential failure of the reactor vessel as a result of postulated PTS events if continued operation beyond the screening criterion is allowed. In the analysis, the licensee may determine the properties of the reactor vessel materials based on available information, research results, and plant surveillance data, and may use probabilistic fracture mechanics techniques. This analysis must be submitted at least three years before $RT_{PTS}$ is projected to exceed the PTS screening criterion.

(5) After consideration of the licensee's analyses, including effects of proposed corrective actions, if any, submitted in accordance with paragraphs (b)(3) and (b)(4) of this section, the Director, Office of Nuclear Reactor Regulation, may, on a case-by-case basis, approve operation of the facility with $RT_{PTS}$ in excess of the PTS screening criterion. The Director, Office of Nuclear Reactor Regulation, will consider factors significantly affecting the potential for failure of the reactor vessel in reaching a decision.

(6) If the Director, Office of Nuclear Reactor Regulation, concludes, pursuant to paragraph (b)(5) of this section, that operation of the facility with $RT_{PTS}$ in excess of the PTS screening criterion cannot be approved on the basis of the licensee's analyses submitted in accordance with paragraphs (b)(3) and (b)(4) of this section, the licensee shall request and receive approval by the Director, Office of Nuclear Reactor Regulation, prior to any operation beyond the criterion. The request must be based upon modifications to equipment, systems, and operation of the facility in addition to those previously proposed in the submitted analyses that would reduce the potential for failure of the reactor vessel due to PTS events, or upon further analyses based upon new information or improved methodology.

(7) If the limiting $RT_{PTS}$ value of the plant is projected to exceed the screening criteria in paragraph (b)(2), or the criteria in paragraphs (b)(3) through (b)(6) of this section cannot be satisfied, the reactor vessel beltline may be given a thermal annealing treatment to recover the fracture toughness of the material, subject to the requirements of §§50.66. The reactor vessel may continue to be operated only for that service period within which the predicted fracture toughness of the vessel beltline materials satisfy the requirements of paragraphs (b)(2) through (b)(6) of this section, with $RT_{PTS}$ accounting for the effects of annealing and subsequent irradiation.

(c) *Calculation of $RT_{PTS}$*. $RT_{PTS}$ must be calculated for each vessel beltline material using a fluence value, f, which is the EOL fluence for the material. $RT_{PTS}$ must be evaluated using the same procedures used to calculate $RT_{NDT}$, as indicated in paragraph (c)(1) of this section, and as provided in paragraphs (c)(2) and (c)(3) of this section.

(1) Equation 1 must be used to calculate values of $RT_{NDT}$ for each weld and plate, or forging, in the reactor vessel beltline.

Equation 1: $RT_{NDT}=RT_{NDT(U)}+M+RT_{NDT}$

(i) If a measured value of $RT_{NDT(U)}$ is not available, a generic mean value for the class(3) of material may be used if there are sufficient test results to establish a mean and a standard deviation for the class.

(ii) For generic values of weld metal, the following generic mean values must be used unless justification for different values is provided: 0°°F for welds made with Linde 80 flux, and -56 °°F for welds made with Linde 0091, 1092 and 124 and ARCOS B-5 weld fluxes.

(iii) *M* means the margin to be added to account for uncertainties in the values of $RT_{NDT(U)}$, copper and nickel contents, fluence and the calculational procedures. M is evaluated from Equation 2.

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(A) In Equation 2, U Image Not Available is the standard deviation for $RT_{NDT(U)}$. If a measured value of $RT_{NDT(U)}$ is used, then Image Not Available is determined from the precision of the test method. If a measured value of $RT_{NDT(U)}$ is not available and a generic mean value for that class of materials is used, then Image Not Available is the standard deviation obtained from the set of data used to establish the mean. If a generic mean value given in paragraph (c)(1)(i)(B) of this section for welds is used, then Image Not Available is 17°°F.

(B) In Equation 2, Image Not Available is the standard deviation for $RT_{NDT}$. The value of Image Not Available to be used is 28°°F for welds and 17°°F for base metal; the value of Image Not Available need not exceed one-half of $RT_{NDT}$.

(iv) Image Not Available $RT_{NDT}$ is the mean value of the transition temperature shift, or change in $RT_{NDT}$, due to irradiation, and must be calculated using Equation 3.

Equation 3: Image Not Available $RT_{NDT}=(CF)f^{(0.28 - 0.10 \log f)}$

A) *CF* (°°F) is the chemistry factor, which is a function of copper and nickel content. CF is given in Table 1 for welds and in Table 2 for base metal (plates and forgings). Linear interpolation is permitted. In Tables 1 and 2, "Wt-% copper" and "Wt-% nickel" are the best-estimate values for the material, which will normally be the mean of the measured values for a plate or forging. For a weld, the best estimate values will normally be the mean of the measured values for a weld deposit made using the same weld wire heat number as the critical vessel weld. If these values are not available, the upper limiting values given in the material specifications to which the vessel material was fabricated may be used. If not available, conservative estimates (mean plus one standard deviation) based on generic data(4) may be used if justification is provided. If none of these alternatives are available, 0.35% copper and 1.0% nickel must be assumed.

(B) *f* is the best estimate neutron fluence, in units of $10^{19}$ n/cm$^2$ (E greater than 1 MeV), at the clad-base-metal interface on the inside surface of the vessel at the location where the material in question receives the highest fluence for the period of service in question. As specified in this paragraph, the EOL fluence for the vessel beltline material is used in calculating $KRT_{PTS}$.

(v) Equation 4 must be used for determining $RT_{PTS}$ using equation 3 with EOL fluence values for determining Image Not Available $RT_{PTS}$.

Equation 4: $RT_{PTS}=RT_{NDT(U)}+M+$ Image Not Available $RT_{PTS}$

(2) To verify that $RT_{NDT}$ for each vessel beltline material is a bounding value for the specific reactor vessel, licensees shall consider plant-specific information that could affect the level of embrittlement. This information includes but is not limited to the reactor vessel operating temperature and any related surveillance program(5) results. (i) Results from the plant-specific surveillance program must be integrated into the $RT_{NDT}$ estimate if the plant-specific surveillance data has been deemed credible as judged by the following criteria:

(A) The materials in the surveillance capsules must be those which are the controlling materials with regard to radiation embrittlement.

(B) Scatter in the plots of Charpy energy versus temperature for the irradiated and unirradiated conditions must be small enough to permit the determination of the 30-foot-pound temperature unambiguously.

(C) Where there are two or more sets of surveillance data from one reactor, the scatter of Image Not Available $RT_{NDT}$ values must be less than 28°°F for welds and 17°°F for base metal. Even if the range in the capsule fluences is large (two or more orders of magnitude), the scatter may not exceed twice those values.

(D) The irradiation temperature of the Charpy specimens in the capsule must equal the vessel wall temperature at the cladding/base metal interface within ±±25°°F.

(E) The surveillance data for the correlation monitor material in the capsule, if present, must fall within the scatter band of the data base for the material.

(ii)(A) Surveillance data deemed credible according to the criteria of paragraph (c)(2)(i) of this section must be used to determine a material-specific value of CF for use in Equation 3. A material-specific value of CF is determined from Equation 5.

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(B) In Equation 5, "n"is the number of surveillance data points, "$A_i$" is the measured value of Image Not Available $RT_{NDT}$ and "$f_i$" is the fluence for each surveillance data point. If there is clear evidence that the copper and nickel content of the surveillance weld differs from the vessel weld, i.e. differs from the average for the weld wire heat number associated with the vessel weld and the surveillance weld, the measured values of Image Not Available $RT_{NDT}$ must be adjusted for differences in copper and nickel content by multiplying them by the ratio of the chemistry factor for the vessel material to that for the surveillance weld.

(iii) For cases in which the results from a credible plant-specific surveillance program are used, the value of Image Not Available to be used is 14°°F for welds and 8.5°°F for base metal; the value of need not exceed one-half of $RT_{NDT}$.

(iv) The use of results from the plant-specific surveillance program may result in an $RT_{NDT}$ that is higher or lower than those determined in paragraph (c)(1).

(3) Any information that is believed to improve the accuracy of the $RT_{PTS}$ value significantly must be reported to the Director, Office of Nuclear Reactor Regulation. Any value of $RT_{PTS}$ that has been modified using the procedures of paragraph (c)(2) of this section is subject to the approval of the Director, Office of Nuclear Reactor Regulation, when used as provided in this section.

Table 1. -- Chemistry Factor for Weld Metals, °°F

Table 2. -- Chemistry Factor for Base Metals, °°F

[60 FR 65468, Dec. 19, 1995, as amended at 61 FR 39300, July 29, 1996]

[2] Changes to $RT_{PTS}$ values are considered significant if either the previous value or the current value, or both values, exceed the screening criterion prior to the expiration of the operating license, including any renewed term, if applicable for the plant.

[3] The class of material for estimating $RT_{NDT(U)}$ is generally determined for welds by the type of welding flux (Linde 80, or other), and for base metal by the material specification.

[4] Data from reactor vessels fabricated to the same material specification in the same shop as the vessel in question and in the same time period is an example of "generic data."

[5] Surveillance program results means any data that demonstrates the embrittlement trends for the limiting beltline material, including but not limited to data from test reactors or from surveillance programs at other plants with or without surveillance program integrated per 10 CFR Part 50, Appendix H.

# Appendix G to Part 50 -- Fracture Toughness Requirements



I. Introduction and Scope

This appendix specifies fracture toughness requirements for ferritic materials of pressure-retaining components of the reactor coolant pressure boundary of light water nuclear power reactors to provide adequate margins of safety during any condition of normal operation, including anticipated operational occurrences and system hydrostatic tests, to which the pressure boundary may be subjected over its service lifetime.

The ASME Code forms the basis for the requirements of this appendix. "ASME Code" means the American Society of Mechanical Engineers Boiler and Pressure Vessel Code. If no section is specified, the reference is to Section III, Division 1, "Rules for Construction of Nuclear Power Plant Components." "Section XI" means Section XI, Division 1, "Rules for Inservice Inspection of Nuclear Power Plant Components." If no edition or addenda are specified, the ASME Code edition and addenda and any limitations and modifications thereof, which are specified in §§50.55a, are applicable.

The sections, editions and addenda of the ASME Boiler and Pressure Vessel Code specified in §§50.55a have been approved for incorporation by reference by the Director of the Federal Register. A notice of any changes made to the material incorporated by reference will be published in the Federal Register. Copies of the ASME Boiler and Pressure Vessel Code may be purchased from the American Society of Mechanical Engineers, United Engineering Center, 345 East 47th Street, New York, NY 10017, and are available for inspection at the NRC Library, 11545 Rockville Pike, Two White Flint North, Rockville, MD 20852 - 2738.

The requirements of this appendix apply to the following materials:

A. Carbon and low-alloy ferritic steel plate, forgings, castings, and pipe with specified minimum yield strengths not over 50,000 psi (345 MPa), and to those with specified minimum yield strengths greater than 50,000 psi (345 MPa) but not over 90,000 psi (621 MPa) if qualified by using methods equivalent to those described in paragraph G - 2110 of Appendix G of Section XI of the latest edition and addenda of the ASME Code incorporated by reference into §§50.55a(b)(2).

B. Welds and weld heat-affected zones in the materials specified in paragraph I.A. of this appendix

C. Materials for bolting and other types of fasteners with specified minimum yield strengths not over 130,000 psi (896 MPa).

Note: The adequacy of the fracture toughness of other ferritic materials not covered in this section must be demonstrated to the Director, Office of Nuclear Reactor Regulation, on an individual case basis.

## II. Definitions

A. *Ferritic material* means carbon and low-alloy steels, higher alloy steels including all stainless alloys of the 4xx series, and maraging and precipitation hardening steels with a predominantly body-centered cubic crystal structure.

B. *System hydrostatic tests* means all preoperational system leakage and hydrostatic pressure tests and all system leakage and hydrostatic pressure tests performed during the service life of the pressure boundary in compliance with the ASME Code, Section XI.

C. *Specified minimum yield strength* means the minimum yield strength (in the unirradiated condition) of a material specified in the construction code under which the component is built under §§50.55a.

D. $RT_{NDT}$ means the reference temperature of the material, for all conditions.

(i) For the pre-service or unirradiated condition, $RT_{NDT}$ is evaluated according to the procedures in the ASME Code, Paragraph NB - 2331.

(ii) For the reactor vessel beltline materials, $RT_{NDT}$ must account for the effects of neutron radiation.

E. Image Not Available $RT_{NDT}$ means the transition temperature shift, or change in $RT_{NDT}$, due to neutron radiation effects, which is evaluated as the difference in the 30 ft-lb (41 J) index temperatures from the average Charpy curves measured before and after irradiation.

. *Beltline* or *Beltline region of reactor vessel* means the region of the reactor vessel (shell material including welds, heat affected zones, and plates or forgings) that directly surrounds the effective height of the active core and adjacent regions of the reactor vessel that are predicted to experience sufficient neutron radiation damage to be considered in the selection of the most limiting material with regard to radiation damage.

## III. Fracture Toughness Tests

A. To demonstrate compliance with the fracture toughness requirements of Section IV of this appendix, ferritic materials must be tested in accordance with the ASME Code and, for the beltline materials, the test requirements of Appendix H of this part. For a reactor vessel that was constructed to an ASME Code earlier than the Summer 1972 Addenda of the 1971 Edition (under §§50.55a), the fracture toughness data and data analyses must be supplemented in a manner approved by the Director, Office of Nuclear Reactor Regulation, to demonstrate equivalence with the fracture toughness requirements of this appendix.

B. Test methods for supplemental fracture toughness tests described in paragraph IV.A.1.b of this appendix must be submitted to and approved by the Director, Office of Nuclear Reactor Regulation, prior to testing.

. All fracture toughness test programs conducted in accordance with paragraphs III.A and III.B must comply with ASME Code requirements for calibration of test equipment, qualification of test personnel, and retention of records of these functions and of the test data.

IV. Fracture Toughness Requirements

A. The pressure-retaining components of the reactor coolant pressure boundary that are made of ferritic materials must meet the requirements of the ASME Code, supplemented by the additional requirements set forth below, for fracture toughness during system hydrostatic tests and any condition of normal operation, including anticipated operational occurrences. Reactor vessels may continue to be operated only for that service period within which the requirements of this section are satisfied. For the reactor vessel beltline materials, including welds, plates and forgings, the values of RTNDT and Charpy upper-shelf energy must account for the effects of neutron radiation, including the results of the surveillance program of Appendix H of this part. The effects of neutron radiation must consider the radiation conditions (i.e., the fluence) at the deepest point on the crack front of the flaw assumed in the analysis.

*1. Reactor Vessel Charpy Upper-Shelf Energy Requirements*

a. Reactor vessel beltline materials must have Charpy upper-shelf energy,(1) in the transverse direction for base material and along the weld for weld material according to the ASME Code, of no less than 75 ft-lb (102 J) initially and must maintain Charpy upper-shelf energy throughout the life of the vessel of no less than 50 ft-lb (68 J), unless it is demonstrated in a manner approved by the Director, Office of Nuclear Reactor Regulation, that lower values of Charpy upper-shelf energy will provide margins of safety against fracture equivalent to those required by Appendix G of Section XI of the ASME Code. This analysis must use the latest edition and addenda of the ASME Code incorporated by reference into §§50.55a(b)(2) at the time the analysis is submitted.

b. Additional evidence of the fracture toughness of the beltline materials after exposure to neutron irradiation may be obtained from results of supplemental fracture toughness tests for use in the analysis specified in section IV.A.1.a.

c. The analysis for satisfying the requirements of section IV.A.1 of this appendix must be submitted, as specified in §§50.4, for review and approval on an individual case basis at least three years prior to the date when the predicted Charpy upper-shelf energy will no longer satisfy the requirements of section IV.A.1 of this appendix, or on a schedule approved by the Director, Office of Nuclear Reactor Regulation.

*2. Pressure-Temperature Limits and Minimum Temperature Requirements*

a. Pressure-temperature limits and minimum temperature requirements for the reactor vessel are given in Table 3, and are defined by the operating condition (i.e., hydrostatic pressure and leak tests, or normal operation including anticipated operational occurrences), the vessel pressure, whether or not fuel is in the vessel, and whether the core is critical. In Table 3, the vessel pressure is defined as a percentage of the preservice system hydrostatic test pressure. The appropriate requirements on both the pressure-temperature limits and the minimum permissible temperature must be met for all conditions.

b. The pressure-temperature limits identified as "ASME Appendix G limits" in Table 3 require that the limits must 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.

c. The minimum temperature requirements given in Table 3 pertain to the controlling material, which is either the material in the closure flange or the material in the beltline region with the highest reference temperature. As specified in Table 3, the minimum temperature requirements and the controlling material depend on the operating condition (i.e., hydrostatic pressure and leak tests, or normal operation including anticipated operational occurrences), the vessel pressure, whether fuel is in the vessel, and whether the core is critical. The metal temperature of the controlling material, in the region of the controlling material which has the least favorable combination of stress and temperature, must exceed the appropriate minimum temperature requirement for the condition and pressure of the vessel specified in Table 1.

d. Pressure tests and leak tests of the reactor vessel that are required by Section XI of the ASME Code must be completed before the core is critical.

B. If the procedures of Section IV.A. of this appendix do not indicate the existence of an equivalent safety margin, the reactor vessel beltline may be given a thermal annealing treatment to recover the fracture toughness of the material, subject to the requirements of §§50.66. The reactor vessel may continue to be operated only for that service period within which the predicted fracture toughness of the beltline region materials satisfies the requirements of Section IV.A. of this appendix using the values of RTNDT and Charpy upper-shelf energy that include the effects of annealing and subsequent irradiation.

**Table 1. - Pressure and Temperature Requirements for the Reactor Pressure Vessel**

| Operating Condition | Vessel Pressure[1] | Requirements for Pressure-Temperature Limits | Minimum Temperature Requirements |
|---|---|---|---|
| 1. Hydrostatic pressure and leak tests (core is not critical): | | | |
| 1.a Fuel in the vessel | ≤20% | ASME Appendix G Limits | (2) |
| 1.b Fuel in the vessel | >20% | ASME Appendix G Limits | (2) +90 °° F(6) |
| 1.c No fuel in the vessel (Preservice Hydrotest Only) | ALL | (Not Applicable) | (3) +60 °° F |
| 2. Normal operation (incl. heat-up and cool-down), including anticipated operational occurrences: | | | |
| 2.a Core not critical | ≤20% | ASME Appendix G Limits | (2) |
| 2.b Core not critical | >20% | ASME Appendix G Limits | (2) + 120 °° F. |
| 2.c Core critical | ≤20% | ASME Appendix G Limits + 40 °° F. | Larger of [(4)] or [(2) + 40°° F.] |
| 2.d Core critical | >20% | ASME Appendix G Limits + 40 °° F. | Larger of [(4)] or [(2)+160°°F] |
| 2.e Core critical for BWR (5) | ≤20% | ASME Appendix G Limits + 40 °° F. | (2)+60°°F |

[1] Percent of the preservice system hydrostatic test pressure.

[2] The highest reference temperature of the material in the closure flange region that is highly stressed by the bolt preload..

[3] The highest reference temperature of the vessel.

[4] The minimum permissible temperature for the inservice system hydrostatic pressure test.

[5] For boiling water reactors (BWR) with water level within the normal range for power operation.

[6] Lower temperatures are permissible if they can be justified by showing that the margins of safety of the controlling region are equivalent to those required for the beltline when it is controlling.

[60 FR 65474, Dec. 19, 1995]

[1] Defined in ASTM E 185 - 79 and - 82 which are incorporated by reference in Appendix H to Part 50.