

Nuclear Regulatory Commission

§ 50.61

(v) Create a possibility for an accident of a different type than any previously evaluated in the final safety analysis report (as updated);

(vi) Create a possibility for a malfunction of an SSC important to safety with a different result than any previously evaluated in the final safety analysis report (as updated);

(vii) Result in a design basis limit for a fission product barrier as described in the FSAR (as updated) being exceeded or altered; or

(viii) Result in a departure from a method of evaluation described in the FSAR (as updated) used in establishing the design bases or in the safety analyses.

(3) In implementing this paragraph, the FSAR (as updated) is considered to include FSAR changes resulting from evaluations performed pursuant to this section and analyses performed pursuant to § 50.90 since submittal of the last update of the final safety analysis report pursuant to § 50.71 of this part.

(4) The provisions in this section do not apply to changes to the facility or procedures when the applicable regulations establish more specific criteria for accomplishing such changes.

(d)(1) The licensee shall maintain records of changes in the facility, of changes in procedures, and of tests and experiments made pursuant to paragraph (c) of this section. These records must include a written evaluation which provides the bases for the determination that the change, test, or experiment does not require a license amendment pursuant to paragraph (c)(2) of this section.

(2) The licensee shall submit, as specified in § 50.4 or § 52.3 of this chapter, as applicable, a report containing a brief description of any changes, tests, and experiments, including a summary of the evaluation of each. A report must be submitted at intervals not to exceed 24 months. For combined licenses, the report must be submitted at intervals not to exceed 6 months during the period from the date of application for a combined license to the date the Commission makes its findings under 10 CFR 52.103(g).

(3) The records of changes in the facility must be maintained until the termination of an operating license

issued under this part, a combined license issued under part 52 of this chapter, or the termination of a license issued under 10 CFR part 54, whichever is later. Records of changes in procedures and records of tests and experiments must be maintained for a period of 5 years.

[64 FR 53613, Oct. 4, 1999, as amended at 66 FR 64738, Dec. 14, 2001; 72 FR 49500, Aug. 28, 2007]

§ 50.60 Acceptance criteria for fracture prevention measures for lightwater nuclear power reactors for normal operation.

(a) Except as provided in paragraph (b) of this section, all light-water nuclear power reactors, other than reactor facilities for which the certifications required under § 50.82(a)(1) have been submitted, must meet the fracture toughness and material surveillance program requirements for the reactor coolant pressure boundary set forth in appendices G and H to this part.

(b) Proposed alternatives to the described requirements in Appendices G and H of this part or portions thereof may be used when an exemption is granted by the Commission under § 50.12.

[48 FR 24009, May 27, 1983, as amended at 50 FR 50777, Dec. 12, 1985; 61 FR 39300, July 29, 1996]

§ 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 Bellline* 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 under this part or a combined license issued under Part 52 of this chapter, other than a nuclear power reactor facility for which the certification required under § 50.82(a)(1) has been submitted, the licensee shall have projected values of RT_{PTS} or RT_{MAX-X} , accepted by the NRC, for each reactor vessel beltline material. For pressurized water nuclear power reactors for which a construction permit was issued under this part before February 3, 2010 and whose reactor vessel was designed and fabricated to the 1998 Edition or earlier of the ASME Code, the projected values must be in accordance

with this section or § 50.61a. For pressurized water nuclear power reactors for which a construction permit is issued under this part after February 3, 2010 and whose reactor vessel is designed and fabricated to an ASME Code after the 1998 Edition, or for which a combined license is issued under Part 52, the projected values must be in accordance with this section. When determining compliance with this section, the assessment of RT_{PTS} must use the calculation procedures described in paragraph (c)(1) and perform the evaluations described 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 significant² 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

²Changes to RT_{PTS} values are considered significant if either the previous value or the current value, or both values, exceed the screening criterion before the expiration of the operating license or the combined license under Part 52 of this chapter, including any renewed term, if applicable for the plant.

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 li-

censee 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.

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

(i) If a measured value of $RT_{NDT(U)}$ is not available, a generic mean value for the class³ of material may be used if there are sufficient test results to establish a mean and a standard deviation for the class.

³The 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.

(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.

$$\text{Equation 2: } M = 2\sqrt{\sigma_U^2 + \sigma_\Delta^2}$$

(A) In Equation 2, σ_U is the standard deviation for $RT_{NDT(U)}$. If a measured value of $RT_{NDT(U)}$ is used, then σ_U 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 σ_U 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 σ_U is 17 °F.

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

(iv) ΔRT_{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.

$$\text{Equation 3: } \Delta RT_{NDT} = (CF)f^{(0.28 - 0.10 \log f)}$$

(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 data⁴ 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/cm² (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 ΔRT_{PTS} .

$$\text{Equation 4: } RT_{PTS} = RT_{NDT(U)} + M + \Delta RT_{PTS}$$

(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 program⁵ 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

⁴Data 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.”

⁵Surveillance 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.

Nuclear Regulatory Commission

§ 50.61

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 ΔRT_{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.

$$\text{Equation 5: } CF = \frac{\sum_{i=1}^n \left[A_i \times f_i^{(0.28-0.10 \log f_i)} \right]}{\sum_{i=1}^n \left[f_i^{(0.56-0.20 \log f_i)} \right]}$$

(B) In Equation 5, “n” is the number of surveillance data points, “ A_i ” is the measured value of ΔRT_{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 ΔRT_{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 σ_A to be used is 14 °F for welds and 8.5 °F for base metal; the value of σ_A need not exceed one-half of ΔRT_{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

| Copper, wt-% | Nickel, wt-% | | | | | | |
|--------------|--------------|------|------|------|------|------|------|
| | 0 | 0.20 | 0.40 | 0.60 | 0.80 | 1.00 | 1.20 |
| 0 | 20 | 20 | 20 | 20 | 20 | 20 | 20 |
| 0.01 | 20 | 20 | 20 | 20 | 20 | 20 | 20 |
| 0.02 | 21 | 26 | 27 | 27 | 27 | 27 | 27 |
| 0.03 | 22 | 35 | 41 | 41 | 41 | 41 | 41 |
| 0.04 | 24 | 43 | 54 | 54 | 54 | 54 | 54 |
| 0.05 | 26 | 49 | 67 | 68 | 68 | 68 | 68 |
| 0.06 | 29 | 52 | 77 | 82 | 82 | 82 | 82 |
| 0.07 | 32 | 55 | 85 | 95 | 95 | 95 | 95 |
| 0.08 | 36 | 58 | 90 | 106 | 108 | 108 | 108 |
| 0.09 | 40 | 61 | 94 | 115 | 122 | 122 | 122 |
| 0.10 | 44 | 65 | 97 | 122 | 133 | 135 | 135 |
| 0.11 | 49 | 68 | 101 | 130 | 144 | 148 | 148 |
| 0.12 | 52 | 72 | 103 | 135 | 153 | 161 | 161 |
| 0.13 | 58 | 76 | 106 | 139 | 162 | 172 | 176 |
| 0.14 | 61 | 79 | 109 | 142 | 168 | 182 | 188 |
| 0.15 | 66 | 84 | 112 | 146 | 175 | 191 | 200 |
| 0.16 | 70 | 88 | 115 | 149 | 178 | 199 | 211 |
| 0.17 | 75 | 92 | 119 | 151 | 184 | 207 | 221 |
| 0.18 | 79 | 95 | 122 | 154 | 187 | 214 | 230 |
| 0.19 | 83 | 100 | 126 | 157 | 191 | 220 | 238 |
| 0.20 | 88 | 104 | 129 | 160 | 194 | 223 | 245 |
| 0.21 | 92 | 108 | 133 | 164 | 197 | 229 | 252 |
| 0.22 | 97 | 112 | 137 | 167 | 200 | 232 | 257 |
| 0.23 | 101 | 117 | 140 | 169 | 203 | 236 | 263 |
| 0.24 | 105 | 121 | 144 | 173 | 206 | 239 | 268 |
| 0.25 | 110 | 126 | 148 | 176 | 209 | 243 | 272 |
| 0.26 | 113 | 130 | 151 | 180 | 212 | 246 | 276 |
| 0.27 | 119 | 134 | 155 | 184 | 216 | 249 | 280 |
| 0.28 | 122 | 138 | 160 | 187 | 218 | 251 | 284 |
| 0.29 | 128 | 142 | 164 | 191 | 222 | 254 | 287 |
| 0.30 | 131 | 146 | 167 | 194 | 225 | 257 | 290 |
| 0.31 | 136 | 151 | 172 | 198 | 228 | 260 | 293 |

§ 50.61a

10 CFR Ch. I (1–1–23 Edition)

TABLE 1—CHEMISTRY FACTOR FOR WELD METALS, °F—Continued

| Copper, wt-% | Nickel, wt-% | | | | | |
|--------------|--------------|------|------|------|------|------|
| | 0 | 0.20 | 0.40 | 0.60 | 0.80 | 1.00 |
| 0.32 | 140 | 155 | 175 | 202 | 231 | 263 |
| 0.33 | 144 | 160 | 180 | 205 | 234 | 266 |
| 0.34 | 149 | 164 | 184 | 209 | 238 | 269 |
| 0.35 | 153 | 168 | 187 | 212 | 241 | 272 |
| 0.36 | 158 | 172 | 191 | 216 | 245 | 275 |
| 0.37 | 162 | 177 | 196 | 220 | 248 | 278 |
| 0.38 | 166 | 182 | 200 | 223 | 250 | 281 |
| 0.39 | 171 | 185 | 203 | 227 | 254 | 285 |
| 0.40 | 175 | 189 | 207 | 231 | 257 | 288 |

TABLE 2—CHEMISTRY FACTOR FOR BASE METALS, °F

| Copper, wt-% | Nickel, wt-% | | | | | |
|--------------|--------------|------|------|------|------|------|
| | 0 | 0.20 | 0.40 | 0.60 | 0.80 | 1.00 |
| 0 | 20 | 20 | 20 | 20 | 20 | 20 |
| 0.01 | 20 | 20 | 20 | 20 | 20 | 20 |
| 0.02 | 20 | 20 | 20 | 20 | 20 | 20 |
| 0.03 | 20 | 20 | 20 | 20 | 20 | 20 |
| 0.04 | 22 | 26 | 26 | 26 | 26 | 26 |
| 0.05 | 25 | 31 | 31 | 31 | 31 | 31 |
| 0.06 | 28 | 37 | 37 | 37 | 37 | 37 |
| 0.07 | 31 | 43 | 44 | 44 | 44 | 44 |
| 0.08 | 34 | 48 | 51 | 51 | 51 | 51 |
| 0.09 | 37 | 53 | 58 | 58 | 58 | 58 |
| 0.10 | 41 | 58 | 65 | 65 | 67 | 67 |
| 0.11 | 45 | 62 | 72 | 74 | 77 | 77 |
| 0.12 | 49 | 67 | 79 | 83 | 86 | 86 |
| 0.13 | 53 | 71 | 85 | 91 | 96 | 96 |
| 0.14 | 57 | 75 | 91 | 100 | 105 | 106 |
| 0.15 | 61 | 80 | 99 | 110 | 115 | 117 |
| 0.16 | 65 | 84 | 104 | 118 | 123 | 125 |
| 0.17 | 69 | 88 | 110 | 127 | 132 | 135 |
| 0.18 | 73 | 92 | 115 | 134 | 141 | 144 |
| 0.19 | 78 | 97 | 120 | 142 | 150 | 154 |
| 0.20 | 82 | 102 | 125 | 149 | 159 | 164 |
| 0.21 | 86 | 107 | 129 | 155 | 167 | 172 |
| 0.22 | 91 | 112 | 134 | 161 | 176 | 181 |
| 0.23 | 95 | 117 | 138 | 167 | 184 | 190 |
| 0.24 | 100 | 121 | 143 | 172 | 191 | 199 |
| 0.25 | 104 | 126 | 148 | 176 | 199 | 208 |
| 0.26 | 109 | 130 | 151 | 180 | 205 | 216 |
| 0.27 | 114 | 134 | 155 | 184 | 211 | 225 |
| 0.28 | 119 | 138 | 160 | 187 | 216 | 233 |
| 0.29 | 124 | 142 | 164 | 191 | 221 | 241 |
| 0.30 | 129 | 146 | 167 | 194 | 225 | 249 |
| 0.31 | 134 | 151 | 172 | 198 | 228 | 255 |
| 0.32 | 139 | 155 | 175 | 202 | 231 | 260 |
| 0.33 | 144 | 160 | 180 | 205 | 234 | 264 |
| 0.34 | 149 | 164 | 184 | 209 | 238 | 268 |
| 0.35 | 153 | 168 | 187 | 212 | 241 | 272 |
| 0.36 | 158 | 173 | 191 | 216 | 245 | 275 |
| 0.37 | 162 | 177 | 196 | 220 | 248 | 278 |
| 0.38 | 166 | 182 | 200 | 223 | 250 | 281 |
| 0.39 | 171 | 185 | 203 | 227 | 254 | 285 |
| 0.40 | 175 | 189 | 207 | 231 | 257 | 288 |

[60 FR 65468, Dec. 19, 1995, as amended at 61 FR 39300, July 29, 1996; 72 FR 49500, Aug. 28, 2007; 73 FR 5722, Jan. 31, 2008; 75 FR 23, Jan. 4, 2010; 84 FR 65644, Nov. 29, 2019]

§ 50.61a Alternate fracture toughness requirements for protection against pressurized thermal shock events.

(a) *Definitions.* Terms in this section have the same meaning as those presented in 10 CFR 50.61(a), with the exception of the term “ASME Code.”

(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,” and Section XI, Division I, “Rules for Inservice Inspection of Nuclear Power Plant Components,” edition and addenda and any limitations and modifications thereof as specified in § 50.55a.

(2) RT_{MAX-AW} means the material property which characterizes the reactor vessel’s resistance to fracture initiating from flaws found along axial weld fusion lines. RT_{MAX-AW} is determined under the provisions of paragraph (f) of this section and has units of °F.

(3) RT_{MAX-PL} means the material property which characterizes the reactor vessel’s resistance to fracture initiating from flaws found in plates in regions that are not associated with welds found in plates. RT_{MAX-PL} is determined under the provisions of paragraph (f) of this section and has units of °F.

(4) RT_{MAX-FO} means the material property which characterizes the reactor vessel’s resistance to fracture initiating from flaws in forgings that are not associated with welds found in forgings. RT_{MAX-FO} is determined under the provisions of paragraph (f) of this section and has units of °F.

(5) RT_{MAX-CW} means the material property which characterizes the reactor vessel’s resistance to fracture initiating from flaws found along the circumferential weld fusion lines. RT_{MAX-CW} is determined under the provisions of paragraph (f) of this section and has units of °F.

(6) RT_{MAX-X} means any or all of the material properties RT_{MAX-AW} , RT_{MAX-PL} , RT_{MAX-FO} , RT_{MAX-CW} , or sum of RT_{MAX-AW} and RT_{MAX-PL} , for a particular reactor vessel.

(7) ϕt means fast neutron fluence for neutrons with energies greater than 1.0 MeV. ϕt is utilized under the provisions of paragraph (g) of this section and has units of n/cm².