4.2 Reactor Vessel Neutron Embrittlement

Review Responsibilities

Primary- Branch responsible for materials and chemical engineering
Secondary - Branch responsible for reactor systems

4.2.1 Areas of Review

The fracture toughness of ferritic steel in the reactor vessel beltline region of light-water nuclear power reactors is reduced during plant service neutron irradiation. Areas of review to ensure that the reactor vessel has adequate fracture toughness to prevent brittle failure during normal and off-normal operating conditions are (1) upper-shelf energy, (2) surveillance program, (3) pressurized thermal shock (PTS) for pressurized water reactors (PWRs), (4) heat-up and cool-down (pressure-temperature limits) curves, and (5) boiling water reactor (BWR) Vessel and Internals Project (VIP) VIP-05 analysis for elimination of circumferential weld inspection for BWRs.

The adequacy of the upper-shelf energy analyses and surveillance programs for light-water reactors, the PTS analyses for pressurized water reactors (PWRs), and the heat-up and cool-down (pressure-temperature limits) curves are reviewed for the period of extended operation.

The branch responsible for reactor systems should review neutron fluence and dosimetry information in the application.

4.2.2 Acceptance Criteria

The acceptance criteria for the areas of review described in Subsection 4.2.1 of this review plan section define acceptable methods for meeting the requirements of the Commission's regulations in 10 CFR 54.21(c)(1).

4.2.2.1 Time-Limited Aging Analysis

Pursuant to 10 CFR 54.21(c)(1)(i) through (iii), an applicant must demonstrate one of the following:

(i) The analyses remain valid for the period of extended operation;

(ii) The analyses have been projected to the end of the extended period of operation; or

(iii) The effects of aging on the intended function(s) will be adequately managed for the period of extended operation.

Specific acceptance criteria for reactor vessel neutron embrittlement depending on the applicant's choice, i.e., 10 CFR 54.21(c)(1)(i), (ii), or (iii), are:

4.2.2.1.1 Upper-Shelf Energy

Paragraph IV.A.1 in Appendix G (Ref. 1) to 10 CFR Part 50 requires that the reactor vessel beltline materials must have Charpy upper-shelf energy of no less than 68 J (50 ft-lb) throughout the life of the reactor vessel unless otherwise approved by NRC.

4.2.2.1.1.1 10 CFR 54.21 (c)(1)(i)

The existing upper-shelf energy analysis remains valid during the period of extended operation because the neutron fluence projected to the end of the period of extended operation is bounded by the fluence assumed in the existing analysis.

4.2.2.1.1.2 10 CFR 54.21(c)(1)(ii)

The upper-shelf energy is re-evaluated to cover the period of extended operation in accordance with Appendix G to 10 CFR Part 50.

4.2.2.1.1.3 10 CFR 54.21(c)(1)(iii)

Acceptance criteria under 10 CFR 54.21(c)(1)(iii) have yet to be developed and will be evaluated on a case-by-case basis to ensure that the aging effects will be managed such that the intended function(s) will be maintained during the period of extended operation.

4.2.2.1.2 Surveillance Program

Appendix H (Ref. 2) to 10 CFR Part 50 requires the reactor vessel materials surveillance program to meet the American Society for Testing and Materials (ASTM) E 185 Standard (Ref. 3). However, the surveillance program in ASTM E 185 is based on plant operation during the current license term, and additional surveillance capsules may be needed for the period of extended operation. Alternatively, an integrated surveillance program for the period of extended operation may be considered for a set of reactors that have similar design and operating features in accordance with Paragraph II.C of Appendix H to 10 CFR Part 50. Additional surveillance capsules may be needed for the period of extended operation for this alternative also.

4.2.2.1.2.1 10 CFR 54.21(c)(1)(i)

Not applicable. As discussed above, the specified surveillance program does not address the period of extended operation.

4.2.2.1.2.2 10 CFR 54.21(c)(1)(ii)

An applicant may provide additional surveillance capsules in its surveillance program.

Specific acceptance criteria for the surveillance program during the period of extended operation have yet to be developed and will be evaluated on a case-by-case basis.

4.2.2.1.2.3 10 CFR 54.21(c)(1)(iii)

The existing reactor vessel material surveillance program should be evaluated for sufficient material data and dosimetry to monitor irradiation embrittlement at the end of the period of extended operation and need for operating restrictions (that is, inlet temperature, neutron spectrum, and flux). If surveillance capsules are not withdrawn during the period of extended operation, operating restrictions should be established to ensure the plant is operated within the environment of the surveillance capsules.

4.2.2.1.3 Pressurized Thermal Shock (for PWRs)

For PWRs, 10 CFR 50.61 (Ref. 4) requires the “reference temperature RT_{PTS}” for reactor vessel beltline materials be less than the “PTS screening criteria” at the expiration date of the operating license unless otherwise approved by NRC. The “PTS screening criteria” are 132°C (270°F) for plates, forgings, and axial weld materials, or 149°(300°F) for circumferential weld materials. The regulations require updating of the pressurized thermal shock assessment upon a request for a change in the expiration date of a facility’s operating license. Therefore, the RT_{PTS} value must be calculated for the reactor life extension period of 48 effective full power years (EFPY).

4.2.2.1.3.1 10 CFR 54.21(c)(1)(i)

The existing PTS analysis remains valid during the period of extended operation because the neutron fluence projected to the end of the period of extended operation is bounded by the fluence assumed in the existing analysis.

4.2.2.1.3.2 10 CFR 54.21(c)(1)(ii)

The PTS analysis is reevaluated to cover the period of extended operation in accordance with 10 CFR 50.61. An analysis is performed in accordance with Regulatory Guide 1.154 (Ref. 5) if the “PTS screening criteria’ in 10 CFR 50.61 are exceeded during the period of extended operation.

4.2.2.1.3.3 10 CFR 54.21(c)(1)(iii)

Acceptance criteria under 10 CFR 54.21(c)(1)(iii) have yet to be developed and will be evaluated on a case-by-case basis to ensure that the aging effects will be managed such that the intended function(s) will be maintained during the period of extended operation.

4.2.2.1.4 Pressure-temperature (P-T) limits

10 CFR Part 50, Appendix G (Ref. 1) requires that heatup and cooldown of the reactor pressure vessel be accomplished within established pressure-temperature (P-T) limits. These limits specify the maximum allowable pressure as a function of reactor coolant temperature. As the reactor pressure vessel becomes embrittled and its fracture toughness is reduced, the allowable pressure is reduced. Operation of the reactor coolant system is also limited by the net positive suction curves for the reactor coolant pumps. These curves specify the minimum pressure required to operate the reactor coolant pumps. Therefore, in order to heatup and cooldown, the reactor coolant temperature and pressure must be maintained within an operating window established between the Appendix G P-T limits and the net positive suction curves.

4.2.2.1.4.1 10 CFR 54.21 (c)(1)(i)

The existing P-T limits are valid during the period of extended operation because the neutron fluence projected to the end of the period of extended operation is bounded by the fluence assumed in the existing analysis.

4.2.2.1.4.2 10 CFR 54.21(c)(1)(ii)

The P-T limits are re-evaluated to cover the period of extended operation in accordance with Appendix G to 10 CFR Part 50 (Ref. 1).

4.2.2.1.4.3 10 CFR 54.21(c)(1)(iii)

An operating window should exist between the P-T limits and the net positive suction curves at the end of the period of extended operation. Appendix G to 10 CFR Part 50 requirements will require periodic update of the P-T limits.

4.2.2.1.5 Elimination of Circumferential Weld Inspection (for BWRs)

Some BWRs have been granted relief from the reactor vessel circumferential shell weld inspections for the current license term because they satisfy the limiting conditional failure probability for the circumferential welds at the expiration of the current license based on BWRVIP 05 and the extent of neutron embrittlement (Refs. 6-8). An applicant for such a BWR may provide justification to extend this relief into the period of extended operation. The staff is currently reviewing BWRVIP-74 which addresses license renewal (Ref. 9). If approved by the staff, BWRVIP-74 may provide the basis for granting such relief.

4.2.2.2 FSAR Supplement

The specific criterion for meeting 10 CFR 54.21(d) is:

The summary description of the evaluation of time-limited aging analyses for the period of extended operation in the FSAR supplement provides appropriate description such that later changes can be controlled by 10 CFR 50.59. The description should contain information associated with the time-limited aging analysis regarding the basis for determining that aging effects are managed in the period of extended operation.

4.2.3 Review Procedures

For each area of review described in Subsection 4.2.1 of this review plan section, the following review procedures are followed:

4.2.3.1 Time-Limited Aging Analysis

For reactor vessel neutron embrittlement, the review procedures, depending on the applicant's choice, i.e., 10 CFR 54.21(c)(1)(i), (ii), or (iii), are:

4.2.3.1.1 Upper-Shelf Energy

4.2.3.1.1.1 10 CFR 54.21(c)(1)(i)

The projected neutron fluence at the end of the period of extended operation is reviewed to verify that it is bounded by the fluence assumed in the existing upper-shelf energy analysis.

4.2.3.1.1.2 10 CFR 54.21(c)(1)(ii)

The revised upper-shelf energy analysis based on the projected neutron fluence at the end of the period of extended operation is reviewed for compliance with Appendix G to 10 CFR Part 50. An applicant may use Regulatory Guide 1.99, Rev. 2 (Ref. 10), to project upper-shelf energy to the end of the period of extended operation. An applicant may also use Appendix K of

Section XI of the ASME Code (Ref. 11) for evaluating upper-shelf energy. The staff should review the applicant's methodology for this evaluation.

4.2.3.1.1.3 10 CFR 54.21(c)(1)(iii)

The applicant's proposal to demonstrate that the effects of aging on the intended function(s) will be adequately managed for the period of extended operation will be reviewed on a case-by-case basis.

4.2.3.1.2 Surveillance Program

4.2.3.1.2.1 10 CFR 54.21(c)(1)(i)

This option is not applicable.

4.2.3.1.2.2 10 CFR 54.21(c)(1)(ii)

The surveillance program is reviewed for its adequacy during the period of extended operation on a case-by-case basis. If an applicant proposes an integrated surveillance program for the period of extended operation for a set of reactors that have similar design and operating features, the proposal is reviewed for compliance with Paragraph II.C of Appendix H to 10 CFR Part 50.

4.2.3.1.2.3 10 CFR 54.21(c)(1)(iii) (Ref. 12)

1. An applicant may project the extent of reactor vessel embrittlement for upper-shelf energy and pressure-temperature limits for 60 years in accordance with Regulatory Guide 1.99, Rev. 2, "Radiation Embrittlement of Reactor Vessel Materials." When using Regulatory Guide 1.99, Rev. 2, an applicant has a choice of the following:

   (a) Neutron Embrittlement Using Chemistry Tables

   An applicant may use the tables in Regulatory Guide 1.99, Rev. 2, to project the extent of reactor vessel neutron embrittlement for the period of extended operation. This is described as Regulatory Position 1 in the Regulatory Guide.

   (b) Neutron Embrittlement Using Surveillance Data

   When credible surveillance data are available, the extent of reactor vessel neutron embrittlement for the period of extended operation may be projected according to Regulatory Position 2 in Regulatory Guide 1.99, rev. 2. The credible data could be collected during the current operating term. The applicant may have a plant-specific program or an integrated surveillance program during the period of extended operation to collect additional data.

2. For an applicant that determines embrittlement using the Regulatory Guide 1.99 tables [see item 1(a) above], the applicant should use the applicable limitations in Regulatory Position 1.3 of the regulatory guide.

3. For an applicant that determines embrittlement using surveillance data [see item 1(b) above], the applicant should define the applicable bounds of the data, such as cold leg

operating temperature and neutron fluence. These bounds should be specific for the referenced surveillance data and would be more restrictive than the bounds for the Regulatory Guide in item 2 above. For example, the plant-specific data could be collected within a smaller temperature range than that in the regulatory guide.

4. All pulled and tested capsules, unless previously discarded, should be placed in storage. (Note: These specimens are saved for future reconstitution use, in case the surveillance program needs to be re-established.)

5. If an applicant has a surveillance program which consists of capsules with a projected fluence of less than the 60-year fluence at the end of 40 years, at least one capsule should remain in the reactor vessel and should be tested during the period of extended operation. The applicant should either delay withdrawal of their last capsule or withdraw a standby capsule during the period of extended operation to monitor the effects of long-term exposure to neutron irradiation.

6. If an applicant has surveillance program which consists of capsules with a projected fluence exceeding the 60-year fluence at the end of 40 years, the applicant should pull these capsules when they reach the 60-year fluence and test one capsule to meet the requirements of ASTM E185 and place the remaining capsules in storage without testing. Any changes in anticipation of additional renewals, however, should be discussed with the staff.

7. Applicants without in-vessel capsules should have alternative dosimetry to monitor neutron fluence during the period of extended operation, as part of the aging management program for reactor vessel neutron embrittlement.

8. The reactor vessel monitoring program should include that, when future plant operations exceed the limitations or bounds in item 2 or 3 above (as applicable) such as operating at a lower cold leg temperature or higher fluence, the impact of plant operation changes regarding the extent of reactor vessel embrittlement will be evaluated and the NRC will be notified. For an applicant without capsules in their reactor vessel, the applicant could propose re-establishing the reactor vessel surveillance program to assess the extent of embrittlement. This program may consist of (1) capsules from item 6 above; (2) reconstitution of specimens from item 4 above; and/or (3) capsules made from any available archival materials. This program could be plant-specific program or an integrated surveillance program.

4.2.3.1.3 Pressurized Thermal Shock (for PWRs)

4.2.3.1.3.1 10 CFR 54.21(c)(1)(i)

The projected neutron fluence at the end of the period of extended operation is reviewed to verify that it is bounded by the fluence assumed in the existing PTS analysis.

4.2.3.1.3.2 10 CFR 54.21(c)(1)(ii)

The revised PTS analysis based on the projected neutron fluence at the end of the period of extended operation is reviewed for compliance with 10 CFR 50.61. There are two methodologies from 10 CFR 50.61 that can be used in the PTS analysis based on the projected neutron fluence at the end of the period of extended operation. RT_{NDT} is the reference temperature (subscript NDT means nil-ductility temperature) used as an indexing parameter to determine the fracture toughness and the amount of embrittlement of a material. RT_{PTS} is the reference temperature used in the PTS analysis and is related to RT_{NDT} at the end of life.

The first methodology does not rely on plant-specific surveillance data to calculate delta RT_{NDT} (i.e., the mean value of the adjustment or shift in reference temperature caused by irradiation). The delta RT_{NDT} is determined by multiplying a chemistry factor from the tables in 10 CFR 50.61 by a fluence factor calculated from the neutron flux using an equation.

The second methodology relies on plant-specific surveillance data to determine the delta RT_{NDT}. In this methodology, two or more sets of surveillance data are needed. Surveillance data consists of a measured delta RT_{NDT} for a corresponding neutron fluence. 10 CFR 50.61 specifies a procedure and a criterion for determining whether the surveillance data are credible, e.g., the difference in the predicted value and the measured value for delta RT_{NDT} must be less than 28°F for weld metal for the surveillance data to be defined as credible. When a credible surveillance data set exists, the chemistry factor determined from the surveillance data can be used in lieu of the values in the table in 10 CFR 50.61 and the standard deviation of the increase in the RT_{NDT} can be reduced from 28°F to 14°F for welds.

If the "PTS screening criteria" in 10 CFR 50.61 are exceeded during the period of extended operation, an analysis based on Regulatory Guide 1.154 is reviewed**.**

4.2.3.1.3.3 10 CFR 54.21(c)(1)(iii)

The applicant's proposal to demonstrate that the effects of aging on the intended function(s) will be adequately managed for the period of extended operation will be reviewed on a case-by-case basis. If the projected reference temperature exceeds the screening criterion established in 10 CFR 50.61, the licensee is required to implement such flux reduction programs as are reasonably practicable to avoid exceeding the screening criterion. The schedule for implementation of such programs may take into account the schedule and anticipated approval by the Director, NRR, of detailed plant-specific analyses to demonstrate acceptable risk with RT_{PTS} above the screening limit. If the licensee cannot avoid exceeding the screening criteria by using a flux reduction program, it must submit a safety analysis to determine what actions are necessary to prevent potential failure of the reactor vessel. 10 CFR 50.61 also permits the licensee to perform a thermal annealing treatment to recover fracture toughness, subject to the requirements of 10 CFR 50.66.

4.2.3.1.4 Pressure-temperature (P-T) limits

4.2.3.1.4.1 10 CFR 54.21(c)(1)(i)

The projected neutron fluence at the end of the period of extended operation is reviewed to verify that it is bounded by the embrittlement assumed in the existing P-T limit analysis.

4.2.3.1.4.2 10 CFR 54.21(c)(1)(ii)

The revised P-T limit analysis based on the projected reduction in fracture toughness at the end of the period of extended operation is reviewed for compliance with 10 CFR Part 50, Appendix G.

4.2.3.1.4.3 10 CFR 54.21(c)(1)(iii)

In order to heatup and cooldown, the reactor coolant temperature and pressure must be maintained within an operating window established between the Appendix G P-T limits and the net positive suction curves. The reviewer verifies that the applicant has provided information to indicate that such an operating window should exist and is sufficient to conduct heatups and cooldowns at the end of the period of extended operation. Appendix G to 10 CFR Part 50 requires periodic update of P-T limits based on projected embrittlement and data from material surveillance program. Thus, the applicant's surveillance program will provide data to update the P-T limits and will manage the reduction in fracture toughness.

4.2.3.1.5 Elimination of Circumferential Weld Inspection (for BWRs)

Some BWRs have been granted relief from the reactor vessel circumferential shell weld inspections for the current license term because they satisfy the limiting conditional failure probability for the circumferential welds at the expiration of the current license based on BWRVIP 05 and the extent of neutron embrittlement (Refs. 6-8). An applicant for such a BWR may provide justification to extend this relief into the period of extended operation. The staff is currently reviewing BWRVIP-74 which supercedes BWRVIP-05 and addresses license renewal (Ref. 9). If approved by the staff, BWRVIP-74 may provide the basis for granting such relief.

When available, an applicant may reference the approved BWRVIP-74 as its basis for requesting the continuation of the relief to the end of the period of extended operation. The staff should review to ensure that the applicant's plant is bounded by the BWRVIP-74 analysis and that the applicant has committed to actions that are the basis for the staff approval of BWRVIP 74.

4.2.3.2 FSAR Supplement

The reviewer verifies that the applicant has provided a FSAR supplement on the summary description of the evaluation of the reactor vessel neutron embrittlement TLAA. Table 4.2-1 of this review plan section contains examples of acceptable FSAR supplement information for this TLAA. The reviewer verifies that the applicant has provided a FSAR supplement using a format similar to that in Table 4.2-1.

4.2.4 Evaluation Findings

The reviewer verifies that sufficient and adequate information has been provided to satisfy the provisions of this review plan section and that the staff's evaluation supports conclusions of the following type depending on the applicant's choice of 10 CFR 54.21(c)(1)(i), (ii), or (iii), to be included in the staff's safety evaluation report.

> The staff evaluation concludes that the applicant has provided an acceptable demonstration, pursuant to 10 CFR 54.21(c)(1), that, for the reactor vessel neutron embrittlement TLAA, (i) the analyses remain valid for the period of extended operation, (ii) the analyses have been projected to the end of the period of extended operation, or (iii) the effects of aging on the intended function(s) will be adequately managed for the period of extended operation. The staff also concludes that the FSAR supplement contains an appropriate summary description of the reactor vessel neutron embrittlement TLAA evaluation for the period of extended operation.

4.2.5 IMPLEMENTATION

Except in those cases in which the applicant proposes an acceptable alternative method, the method described herein will be used by the staff in its evaluation of conformance with Commission regulations.

4.2.6 References

1. 10 CFR Part 50 Appendix G, “Fracture Toughness Requirements.”

2. 10 CFR Part 50, Appendix H, “Reactor Vessel Material Surveillance Program Requirements.”

3. ASTM E 185, “Standard Practice of Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels,” American Society for Testing and Materials, 1982.

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

5. Regulatory Guide 1.154, “Format and Content of Plant-Specific Pressurized Thermal Shock Safety Analysis Reports for Pressurized Water Reactors,” January 1987.

6. BWRVIP-05, “BWR Vessel and Internals Project, BWR Reactor Pressure Vessel Shell Weld Inspection Recommendations (BWRVIP-05),” Boiling Water Reactor Owners Group, September 28, 1995.

7. Letter to Carl Terry of Niagara Mohawk Power Company, from Gus C. Lainas of NRC, dated July 28, 1998.

8. Generic Letter 98-05, “Boiling Water Reactor Licensees Use of the BWRVIP-05 Report to Request Relief from Augmented Examination Requirements on Reactor Pressure Vessel Circumferential Shell Welds,” Nuclear Regulatory Commission, November 10, 1998.

9. BWRVIP-74, “BWR Reactor Pressure Vessel Inspection and Flaw Evaluation Guidelines,” Boiling Water Reactor Owners Group.

10. Regulatory Guide 1.99 Rev. 2, “Radiation Embrittlement of Reactor Vessel Materials,” May, 1988.

11. Appendix K of ASME Code, Section XI, “Rules for Inservice Inspection of Nuclear Power Plant Components.”

12. Letter from C. I. Grimes (NRC) to D. J. Walters (NEI), License Renewal Issue No. 98-0085, “Reactor Vessel Surveillance Program,” dated Dec 3, 1999.

Table 4.2-1. Examples of FSAR Supplement for Reactor Vessel Neutron Embrittlement TLAA Evaluation

TLAA	Description of Evaluation	Implementation Schedule
Upper-shelf energy	Paragraph IV.A.1 in Appendix G to 10 CFR Part 50 requires that the reactor vessel beltline materials must have Charpy upper-shelf energy of no less than 50 ft-lb throughout the life of the reactor vessel unless otherwise approved by the NRC. The upper-shelf energy has been determined to exceed 50 ft-lb to the end of the period of extended operation.	Completed
Surveillance program	Irradiating and testing of metallurgical samples are used to monitor the progress of neutron embrittlement as a function of neutron fluence. The current program is in accordance with ASTM E 185. The program consists of 6 capsules in each unit, with 2 capsules tested, 3 capsules to be tested, and one standby capsule. The withdrawal schedule will be revised to provide data at neutron fluence equal to or greater than the projected peak fluence at the end of the license renewal period. If the last capsule is withdrawn before year 55, will establish reactor vessel neutron environment conditions applicable to the surveillance data. If the plant operates outside of the limits established by these conditions, will inform the NRC and determine the impact of the condition on reactor vessel integrity. If the last capsule is withdrawn before year 55, will install neutron dosimetry to permit tacking of the fluence to the reactor vessel.	The surveillance capsule withdrawal schedule will be revised by....
Pressurized thermal shock (for PWRs)	For PWRs, 10 CFR 50.61 requires the "reference temperature RT_{PTS}" for reactor vessel beltline materials be less that the "PTS screening criteria" at the expiration date of the operating license unless otherwise approved by the NRC. The "PTS screening criteria" are 270 ^{0}F for plates, forgings, and axial weld materials, or 300 ^{0}F for circumferential weld materials. The "reference temperature" has been determined to be less than the "PTS screening criteria" at the end of the period of extended operation.	Completed
Pressure-temperature (P-T) limits	Appendix G to 10 CFR Part 50 requires that heatup and cooldown of the reactor pressure vessel be accomplished within established P-T limits. These limits specify the maximum allowable pressure as a function of	Update as required by Appendix G to 10 CFR Part 50

	reactor coolant temperature. As the reactor pressure vessel becomes embrittled and its fracture toughness is reduced, the allowable pressure is reduced. Appendix G to 10 CFR Part 50 requires periodic update of P-T limits based on projected embrittlement and data from material surveillance program.	
Elimination of circum-ferential weld inspection (for BWRs)	NRC has granted relief from the reactor vessel circumferential shell weld inspections, because the plant has been demonstrated to meet BWRVIP-74 as approved by the NRC.	Completed