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

During plant service, neutron irradiation reduces the fracture toughness of ferritic steel in the reactor vessel beltline region of light-water nuclear power reactors. 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) pressurized thermal shock (PTS) for pressurized water reactors (PWRs), (3) heat-up and cool-down (pressure-temperature limits) curves, and (4) boiling water reactor (BWR) Vessel and Internals Project (VIP) VIP-05 analysis for elimination of circumferential weld inspection and analysis of the axial welds.

The adequacy of the upper-shelf energy analyses for light-water reactors, the PTS analyses for pressurized water reactors (PWRs), the heat-up and cool-down (pressure-temperature limits) curves, and BWR reactor vessel circumferential and axial welds 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 depend 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 a Charpy upper-shelf energy of no less than 68 J (50 ft-lb) throughout the life of the reactor vessel unless otherwise approved by the 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 consider 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 Pressurized Thermal Shock (for PWRs)

For PWRs, 10 CFR 50.61 (Ref. 2) requires that 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 the NRC. The "PTS screening criteria" are 132°C (270°F) for plates, forgings, and axial weld materials, or 149°C(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 entire life of the facility, including the period of extended operation.

4.2.2.1.2.1 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.

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

The PTS analysis is reevaluated to consider the period of extended operation in accordance with 10 CFR 50.61. An analysis is performed in accordance with Regulatory Guide 1.154 (Ref. 3) if the "PTS screening criteria" in 10 CFR 50.61 are exceeded during the period of extended operation.

4.2.2.1.2.3 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.

4.2.2.1.3 Pressure-temperature (P-T) limits

10 CFR Part 50, Appendix G (Ref. 1) requires that the reactor pressure vessel be maintained within established pressure-temperature (P-T) limits including during heatup and cooldown. 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 (given the required minimum temperature) is reduced.

4.2.2.1.3.1 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.

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

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

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

Not applicable. Updated P-T limits for the period of extended operation must be available prior to entering the period of extended operation. (It is not necessary to implement P-T limits to carry the RV through 60 years at the time of application. The updated limits must merely be available prior to the period of extended operation.)

4.2.2.1.4 Elimination of Circumferential Weld Inspection (for BWRs)

Some BWRs have an approved technical alternative eliminating 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. 4-6). An applicant for renewal of a license to operate 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 this issue in the context of license renewal (Ref. 7). Section A.4.5 of Report BWRVIP-74 indicates that the staff's safety evaluation report (SER) conservatively evaluated BWR RPV's to 64 effective full power years (EFPY), which is 10 EFPY greater than what is realistically expected for the end of the license renewal period. Since this was a generic analysis, the licensee must provide plant-specific information to demonstrate that the circumferential beltline weld materials meet the criteria specified in the report and operator training and procedures will be utilized during the license renewal term to limit the frequency for cold over-pressure events.

4.2.2.1.5 Axial Welds (for BWRs)

The staff's SER contained in a letter to Carl Terry dated March 7, 2000, (Ref. 8) discussed the staff's concern related to RPV failure frequency for axial welds and the BWRVIP's analysis of the RPV failure frequency of axial welds. The SER indicates that the RPV failure frequency due to failure of the limiting axial welds in the BWR fleet at the end of 40 years of operation are below 5×10^{-6} per reactor year, given the assumptions on flaw density, distribution and location described in the SER. Since the BWRVIP analysis was generic, the licensee must provide plant-specific information to demonstrate that the axial beltline weld materials meet the criteria specified in the report and operator training and procedures will be utilized during the license renewal term to limit the frequency for cold over-pressure events.

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 analyses regarding the basis for determining that the applicant has made the demonstration required by 10 CFR 54.21(c)(1).

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 documented results of 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. 9), 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. 10) 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 Pressurized Thermal Shock (for PWRs)

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

The documented results of 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.

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

The documented results of 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, i.e., 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.

4.2.3.1.2.3 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.

4.2.3.1.3 Pressure-temperature (P-T) limits

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

The documented results of 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.

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

The documented results of 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.

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

Not applicable.

4.2.3.1.4 Elimination of Circumferential Weld Inspection (for BWRs)

To demonstrate that the vessel has not been embrittled beyond the basis for the technical alternative and that cold over pressure events are not likely to occur during the license renewal term, the applicant should provide: (1) a comparison of the neutron fluence, initial RT_{NDT} , Chemistry Factor amounts of copper and nickel, delta RT_{NDT} and mean RT_{NDT} of the limiting circumferential weld at the end of license renewal period to the 64 EFPY reference case in Appendix E of the staff's SER, (2) an estimate of conditional failure probability of the RPV at the end of the license renewal term based on the comparison of the Mean RT_{NDT} for the limiting circumferential welds and the reference case, and (3) identify procedures and training that will be utilized during the license renewal term to limit the frequency of cold over pressure events to the amount specified in the staff's SER. The staff should ensure that the applicant's plant is bounded by the BWRVIP analysis and that the applicant has committed to actions that are the basis for the staff approval.

4.2.3.1.5 Axial Welds (for BWRs)

To demonstrate that the vessel has not been embrittled beyond the basis for the staff and BWRVIP analyses, the applicant should provide: (1) a comparison of the neutron fluence, initial RT_{NDT} , Chemistry Factor amounts of copper and nickel, delta RT_{NDT} and mean RT_{NDT} of the limiting axial weld at the end of license renewal period to the reference case in the BWRVIP and staff analyses and (2) an estimate of conditional failure probability of the RPV at the end of the license renewal term based on the comparison of the mean RT_{NDT} for the limiting axial welds and the reference case. If this comparison does not indicate that the RPV failure frequency for axial welds is less than 5×10^{-6} per reactor year, provide a probabilistic analysis to determine the RPV failure frequency for axial welds. The staff should ensure that the applicant's plant is bounded by the BWRVIP analysis and that the applicant has committed to actions that are the basis for the staff approval.

4.2.3.2 FSAR Supplement

The reviewer verifies that the applicant has provided information to be included in the FSAR supplement including a 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 with information equivalent to that in Table 4.2-1. The staff expects to impose a license condition in the renewed license, if granted, to require the applicant to update its FSAR to include this FSAR supplement at the next update required pursuant to 10 CFR 50.71(e)(4). As part of the license condition, until the FSAR update is complete, the applicant may make changes to the programs described in its FSAR supplement without prior Commission approval, provided that the applicant evaluates each such change pursuant to the criteria set forth in 10 CFR 50.59.

As noted in Table 4.2-1, an applicant need not incorporate the implementation schedule into its FSAR. However, an applicant should identify and commit to any future aging management activities to be completed before the period of extended operation. The staff expects to impose a license condition in the renewed license, if granted, to ensure that the applicant will complete these activities no later than the committed date.

4.2.4 Evaluation Findings

The reviewer verifies that the applicant has provided sufficient information 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 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 [choose what is appropriate], (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 50.61, “Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events.”

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

4. 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.

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

6. 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.

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

8. Letter to Carl Terry of Niagara Mohawk Power Company, from Jack R. Strosnider, Jr., of NRC, dated March 7, 2000.

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

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

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
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 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. The P-T limits will be updated to consider the period of extended operation.	Update should be completed before the period of extended operation.
Elimination of circum-ferential weld inspection and analysis of axial welds (for BWRs)	NRC has granted relief from the reactor vessel circumferential shell weld inspections, because the applicant has demonstrated through plant-specific analysis that the plant meets BWRVIP-74 as approved by the NRC and has provided sufficient information that the probability of vessel failure due to embitterment of axial welds is low.	Completed

*An applicant need not incorporate the implementation schedule into its FSAR. However, an applicant should identify and commit to any future aging management

activities to be completed before the period of extended operation. The staff expects to impose a license condition in the renewed license, if granted, to ensure that the applicant will complete these activities no later than the committed date.