SUPPLEMENTAL RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

APR1400 Design Certification

Korea Electric Power Corporation / Korea Hydro & Nuclear Power Co., LTD

Docket No. 52-046

RAI No.:	**293-8332**
SRP Section:	**04.03 - Nuclear Design**
Application Section:	**Neutron Fluence Calculation Methodology TeR Sections 2, 3.1, 5 and DCD Table 1.8-2, Sections 5.3.1.6.4.1, 5.3.4**
Date of RAI Issue:	**11/05/2015**

Question No. 04.03-6

Question 4.3-6: Vessel fluence analysis and surveillance

REQUIREMENTS AND GUIDANCE

The requirements pertaining to vessel fluence analysis and surveillance are as follow:

- 10 CFR Part 50 Appendix A, General Design Criterion (GDC) 14 as it relates to ensuring an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture of the reactor coolant pressure boundary, in part, insofar as it considers calculations of neutron fluence.
- GDC 31 as it relates to ensuring that the reactor coolant pressure boundary will behave in a nonbrittle manner and that the probability of rapidly propagating fracture is minimized, in part, insofar as it considers calculations of fluence.
- Appendix G, to 10 CFR Part 50, as it relates to reactor pressure vessel material fracture toughness requirements, in part, insofar as it considers calculations of neutron fluence.
- Appendix H, to 10 CFR Part 50, as it relates to reactor pressure vessel material surveillance program requirements, in part, insofar as it considers calculations of neutron fluence.
- 10 CFR 50.61 as it relates to fracture toughness criteria for PWRs relevant to pressurized thermal shock events, in part, insofar as it considers calculations of neutron fluence.

SRP 4.3 and SRP 5.3 guide the reviewer to apply the following acceptance criteria:

- There is reasonable assurance that the proposed design limits can be met for the expected range of reactor operation, taking into account analysis uncertainties.
- There is reasonable assurance that during normal operation the design limits will not be exceeded.

- The acceptance criteria of Regulatory Guide 1.190, “Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence.”
- The acceptance criteria of Regulatory Guide 1.99, “Radiation Embrittlement of Reactor Vessel Materials.”

ISSUES

(a) The staff reviewed the nuclide compositions of the modeled reactor materials and the fission spectra and neutron source spectrum data provided respectively in Tables 2-1 and 2-2 of the applicant’s technical report by performing hand calculations and comparing the source spectrum with published independent data. The staff found no erroneous data but was not able to discern and confirm the nuclide fractions the applicant uses for mixing the fission source spectra.
(b) The staff’s review noted that the technical report does not identify the numerical options (e.g., differencing schemes) used in the applicant’s DORT code fluence calculations.
(c) The staff’s review compared the calculated-to-measured (C/M) data reported by the applicant with those reported in a published VENUS-1 benchmarking study and confirmed that they are mutually consistent. The technical report indicates that the applicant has used the C/M ratios to determine the bias in the calculated vessel fluence. However, the report does not explain how the reported total bias of 6 percent was calculated.
(d) The applicant’s technical report includes a brief description of the use of in-vessel surveillance capsules. The staff however was not able to discern from the report the axial locations of the capsules.
(e) The staff reviewed the calculated flux and fluence values presented in Table 5.1 and Figures 5.1 and 5.2 of the applicant’s technical report and found them to show relative behaviors consistent with those reported for similar designs in independent published studies. The staff is concerned, however, that the report does not state the fluence limits for the APR1400 vessel. Because of this, the staff cannot discern whether the calculated peak fluence for 55.8 effective full power years would meet the limits for this design.

INFORMATION NEEDED

To address the above concerns, the applicant should provide the following information and update the affect parts of the DCD and its incorporated references accordingly:

a) The fission nuclide mixing fractions used in calculating the neutron source spectrum
b) The DORT code numerical options used for the vessel fluence calculations
c) The method used for determining vessel fluence bias value of 6 percent
d) The axial locations of the in-vessel surveillance capsules
e) The vessel fluence limits in relation to the calculated peak fluences.

Response

a) The used fractions for mixing the fission source spectra are the cycle average relative fission rate as presented in Table 1.

Table 1 Physics Data and Relative Fission Rates for APR1400

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The neutron source spectrum ($S(E)$) was determined by combining the volumetric fission density (c_0) with the relative fission rates (f_i), number of neutrons per fission (ν_i), and fission spectrums (χ_i) for each fissionable nuclide as

$$S(E) = c_0 \times \sum_i f_i \times \nu_i \times \chi_i(E).$$

b) Main DORT options used for vessel fluence calculation are as follows:

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c) The bias values in Table 3-3 in the neutron fluence calculation methodology TeR are not calculated-to-measured (C/M) values but measured-to-calculated (M/C) values. The description (C/M) in the footnote of Table 3.3 is an editorial error and will be corrected. The value for neutron pad in Table 3-3 is used for the vessel fluence bias (+6%) which is from 1.057 (≈1.06). 1.057 has the meaning that the measured value is approximately 6% greater than the calculated one.

d) The axial length of surveillance capsule assembly is approximately 114 inches and the mid-plane of surveillance capsule assembly coincides with the mid-plane of the active core.

e) There is no design limit for the vessel fluence value itself. The integrity of reactor pressure vessel is examined by evaluating the Reference Temperature for Nil Ductility Transition (RT_{NDT}) which uses the neutron fluence as an input.

Supplemental Response

The DCD Table 1.8-2, Section 5.3.1.6.4.1 Flux Measurements, and Section 5.3.4 Combined License Information will be revised to state that a COL applicant is to provide plant-specific surveillance capsule data to benchmark APR1400-Z-A-NR-14015-NP and demonstrate applicability to the specific plant.

Impact on DCD

DCD Table 1.8-2, Sections 5.3.1.6.4.1 and 5.3.4 will be revised to include COL item as indicated in the Attachment.

Impact on PRA

There is no impact on the PRA.

Impact on Technical Specifications

There is no impact on the Technical Specifications.

Impact on Technical/Topical/Environmental Reports

The footnote of Table 3.3 in the neutron fluence calculation methodology TeR will be revised as indicated in the Attachment.

Table 3-3 Bias of Each Region for Equivalent Fission Flux

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Table 1.8-2 (8 of 29)

Item No.	Description
COL 5.2(8)	The COL applicant is to provide and develop the implementation milestones for the inservice inspection and testing program for the RCPB, in accordance with ASME Code Section XI and 10 CFR 50.55a.
COL 5.2(9)	The COL applicant is to address the provisions to accessibility of Class 1 components for ISI if the design of the APR1400 Class 1 component is changed from the DCD design.
COL 5.2(10)	The COL applicant is to provide the list of Code exemptions in the ISI program of the specific plants, if it exists.
COL 5.2(11)	The COL applicant is to prepare and provide any requests for relief from the ASME Code requirements that are impracticable as a result of limitations of component design, geometry, or materials of construction for the specific plants, if necessary. The request will contain the information on applicable Code requirements, alternative ISI method, and justification.
COL 5.2(12)	The COL applicant may invoke ASME Code Cases listed in NRC RG 1.147 for the ISI program.
COL 5.2(13)	The COL applicant is to prepare and implement a boric acid corrosion (BAC) prevention program compliant with Generic Letter 88-05.
COL 5.2(14)	The COL applicant is to prepare the preservice inspection and testing program.
COL 5.2(15)	The COL applicant is to address and develop milestones for preparation and implementation of the procedure for operator responses to prolonged low level leakage.
COL 5.3(1)	The COL applicant is to provide a reactor vessel material surveillance program for a specific plant.
COL 5.3(2)	The COL applicant is to develop P-T limit curves based on plant-specific data.
COL 5.3(3)	The COL applicant is to verify the RT_{PTS} value and the USE at EOL based on plant-specific material property and neutron fluences.
COL 5.3(4)	The COL applicant is to provide and develop the inservice inspection and testing program for the RCPB, in accordance with ASME Section XI and 10 CFR 50.55a.
COL 5.4(1)	The COL applicant is to prepare operational procedures and maintenance programs as related to leak detection and contamination control of RCS.
COL 5.4(2)	The COL applicant is to maintain complete documentation of system design, construction, design modifications, field changes, and operations of RCS.
COL 5.4(3)	The COL applicant is to prepare operational procedures and maintenance programs as related to leak detection and contamination control of SCS.
COL 5.4(4)	The COL applicant is to maintain complete documentation of system design, construction, design modifications, field changes, and operations of SCS.
COL 5.4(5)	The COL applicant is to verify the as-built RV support material properties and 60-year neutron fluence.

Add
COL 5.3(5),
The COL applicant is to provide plant-specific surveillance capsule data to benchmark APR1400-Z-A-NR-14015-P and demonstrate applicability to the specific plant.

Cadmium covers are used for the materials with competing neutron capture activities. The flux monitors are placed in holes drilled in stainless steel housings, as shown in Figure 5.3-3, at three axial locations in each capsule assembly to provide an axial profile of the level of fluence that the specimens attain.

To determine the neutron fluence at the inner surface of the reactor vessel both of the measured value from detectors and the calculated value by transport theory are considered in accordance with NRC RG 1.190. Calculation methods for the neutron flux and fluence are described in Subsection 4.3.3.3.

Add "The COL applicant is to provide plant-specific surveillance capsule data to benchmark APR1400-Z-ANR-14015-P and demonstrate applicability to the specific plant(COL 5.3(5))."

5.3.1.6.4.2 Temperature Estimates

Because the changes in mechanical and i
highly dependent on the irradiation tempera
specimens as well as the pressure vessel. During irradiation, instrumented capsules are not practicable for a surveillance program extending over the design lifetime of a power reactor. The maximum temperature of the irradiated specimens can be estimated with reasonable accuracy by including small pieces of low melting point alloys or pure metals in the capsule assemblies. The compositions of candidate materials with melting points in the operating range of power reactors are listed in Table 5.3-6. The monitors are selected to bracket the operating temperature of the reactor vessel. The temperature monitors consist of a helix of low melting alloy wire inside a sealed quartz tube. A stainless steel weight is provided to destroy the integrity of the wire when the melting point of the alloy is reached. The compositions and the melting temperatures of the temperature monitors are differentiated by the physical lengths of the quartz tubes that contain the alloy wires.

A set of temperature monitors is included in each capsule assembly. The temperature monitors are placed in holes drilled in stainless steel housings, as shown in Figure 5.3-3, and provide the maximum temperature to which the specimens are exposed.

5.3.1.6.5 Irradiation Locations

The test specimens are enclosed within six capsule assemblies. The axial positions of capsule assemblies are bisected by the midplane of the core as shown in Figure 5.3-6. A summary of the specimens contained in each of these capsule assemblies is presented in

COL 5.3(4) The COL applicant is to provide and develop the inservice inspection and testing program for the RCPB, in accordance with ASME Section XI and 10 CFR 50.55a.

Add
COL 5.3(5) The COL applicant is to provide plant-specific surveillance capsule data to benchmark APR1400-Z-A-NR-14015-P and demonstrate applicability to the specific plant.

5.3.5 References

1. 10 CFR Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants," U.S. Nuclear Regulatory Commission.

2. 10 CFR 50.55a, "Codes and Standards," U.S. Nuclear Regulatory Commission.

3. 10 CFR 50.60, "Acceptance Criteria for Fracture Prevention Measures for Light Water Nuclear Power Reactors for Normal Operation" U.S. Nuclear Regulatory Commission.

4. 10 CFR Part 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," U.S. Nuclear Regulatory Commission.

5. 10 CFR Part 50, Appendix G, "Fracture Toughness Requirements," U.S. Nuclear Regulatory Commission.

6. 10 CFR Part 50, Appendix H, "Reactor Vessel Material Surveillance Program Requirements," U.S. Nuclear Regulatory Commission.

7. NUREG-0800, Standard Review Plan, BTP 5-3, "Fracture Toughness Requirements," Rev. 2, U.S. Nuclear Regulatory Commission, March 2007.

8. ASME Boiler and Pressure Vessel Code, Section III, "Rules for Construction of Nuclear Facility Components," The American Society of Mechanical Engineers, the 2007 Edition with the 2008 Addenda.

9. Regulatory Guide 1.50, "Control of Preheat Temperature for Welding of Low-Alloy Steel," Rev. 1, U.S. Nuclear Regulatory Commission, March 2011.

10. ASME Boiler and Pressure Vessel Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," The American Society of Mechanical Engineers, the 2007 Edition with the 2008 Addenda.

11. Regulatory Guide 1.31, "Control of Ferrite Content in Stainless Steel Weld Metal," Rev. 4, U.S. Nuclear Regulatory Commission, October 2013.