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:	**4.3**
Date of RAI Issue:	**11/05/2015**

Question No. 04.03-5

REQUIREMENTS AND GUIDANCE

General Design Criterion (GDC) 10 requires the reactor design to include appropriate margin to assure that specified acceptable fuel design limits (SAFDLs) are not exceeded during normal operation or anticipated operational occurrences (AOOs). GDC 20, “Protection System Functions,” requires automatic initiation of the reactivity control systems to assure that SAFDLs are not exceeded as a result of AOOs and that automatic operation of systems and components important to safety occurs under accident conditions. In addition, GDC 28, “Reactivity Limits,” requires that the effects of postulated reactivity accidents neither result in damage to the reactor coolant pressure boundary greater than limited local yielding nor cause sufficient damage to impair significantly the capability to cool the core. All of these requirements involve accurate knowledge of the total and differential reactivity worths of the control element assemblies (CEAs).

SRP Section 4.3 guides the reviewer to address the following: “The adequacy of the control systems to assure that the reactor can be returned to and maintained in the cold shutdown condition at any time during operation. The applicant shall discuss shutdown margins. Shutdown margins need to be demonstrated by the applicant throughout the fuel cycle.”

ISSUE

The applicant’s nuclear design includes full-strength and part-strength CEAs. The full-strength CEAs use B_4C as neutron absorber. On page 4.3-7, the DCD states: “Methods of controlling the power distribution include the use of full- or part-strength CEAs to alter the axial power distribution; decreasing CEA insertion by boration, thereby improving the radial power distribution; and correcting off-optimum conditions that cause margin degradations such as CEA misoperation.” On page 4.3-17, the DCD also indicates: “The regulating CEA groups can be used to compensate for changes in reactivity associated with routine power level changes.

In addition, they can be used to compensate for minor variations in moderator temperature and boron concentration during operation at power and to dampen axial xenon oscillations." On page 4.3-24, the DCD states: "Control action with part-strength rods or full-strength rods may be required to limit the magnitude of the oscillation." As such, some or all of the regulating rods may be inserted into the core for extended periods of time and at various depths during power operations. The staff is therefore concerned that progressive poison burnout in the affected regulating rods could unacceptably degrade their effectiveness for control and safe shutdown.

On June 23, 2015, staff issued RAI No. 48-7943, Question 4.03-2, requesting specific information to address this concern. On July 22, 2015, ADAMS Accession No. ML15203A536, the applicant responded with the requested information. In addition to computed estimates showing substantial peak B-10 burnout near the CEA tips, the response also included a summary of physics test results from operating OPR1000 cores that are similar in this regard to APR1400. The latter showed that total rod bank worths measured after multiple operating OPR1000 fuel cycles still matched predictions within allowed uncertainty limits even though the predictions neglected B-10 depletion. The staff also audited related calculation notes provided by the applicant that further detail the applicant's bases for the provided B-10 burnout estimates.

The staff notes that the reported insertion histories of the physics-tested OPR1000 regulating rods were significantly less than assumed in the provided burnout estimates. The staff further notes that the reported B-10 burnout estimates were substantial even though they were based on CEA insertion history assumptions (i.e., with regulating full-strength CEAs partly inserted for a total of only 8 months over 10 years of operation at 87 percent capacity) that were not shown to be conservative or bounding. Accordingly, the staff is concerned that the effects of B-10 burnout on CEA worths could in fact become unacceptable unless prevented by the specification of appropriate CEA service limits. Such limits could be placed for example on either CEA replacement intervals or allowed CEA insertion histories.

INFORMATION NEEDED

The applicant is requested to either (a) specify CEA service limits as necessary to prevent unacceptable levels of B-10 burnout in full-strength CEAs used as regulating rods, or else (b) demonstrate that no such CEA service limits are necessary.

Response

The staff noted that the assumed insertion history, as described in Table 1 of RAI No. 48-7943, was not shown to be conservative or bounding. However, the assumed insertion history of the CEAs is considered to be sufficiently conservative since all CEA banks including shutdown bank were 28% inserted in the B-10 burnout evaluation on control rod worth even though only the lead bank (bank 5) are permitted for insertion in the core at HFP during the operation. For more information, the measured regulating CEA insertion histories (mainly bank 5) were shown for HANBIT unit 3 illustrated by Figure 1. In the figure, the accumulated insertion time of regulating bank 5 during 10 years between the 1^{st} and 2^{nd} replacements of all CEAs (Cycle 9 through 15) were [ ]^{TS} which is less than half of assumed insertion histories in the B-10 burnout estimation.

Also, the effect of B-10 burnout estimates on control rod worth was a concern in this issue. This concern could be resolved by the fact that the effect of B-10 burnout is implicitly included in the rod worth uncertainty, and there is no impact on safety assessment as described in RAI No. 48-7943. Furthermore, execution of rod worth measurement is required before startup of each cycle to determine if the shutdown margin and each rod worth are consistent with the design predictions. This test confirms that the B-10 burnout on CEA worth could be acceptable. Accordingly, no additional CEA service limit is necessary for B-10 burnout effect.

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Figure 1 Regulating CEA Insertion History of HANBIT Unit 3

Impact on DCD

There is no impact on DCD.

Impact on PRA

There is no impact on PRA.

Impact on Technical Specifications

There is no impact on Technical Specifications.

Impact on Technical/Topical/Environmental Reports

There is no impact on any Technical, Topical, or Environment Report.

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Docket No. 52-046

RAI No.:	**293-8332**
SRP Section:	**04.03 - Nuclear Design**
Application Section:	**Neutron Fluence Calculation Methodology TeR Sections 2, 5, and Subsection 3.1**
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 ($S(\mathrm{E})$) 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.

Impact on DCD

There is no impact on the DCD.

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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