REVISED 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.:	**291-8347**
SRP Section:	**04.05.02 – Reactor Internal and Core Support Structure Materials**
Application Section:	**4.5.2**
Date of RAI Issue:	**11/04/2015**

Question No. 04.05.02-5

APR-1400 FSAR Section 4.5.2.6 details evaluations conducted concerning irradiation-assisted stress corrosion cracking (IASCC) and void swelling. Significant detail is provided concerning the calculation of neutron fluence and temperature. The staff requires additional information regarding the specific criteria used to evaluate the components for IASCC and void swelling to make a safety finding.

The staff requests that the applicant provide the report(s) that evaluate the reactor internals and core support structures using specific criteria including neutron fluence values, stress values, and temperature. Alternatively that the applicant revise FSAR Section 4.5.2.6 to include a complete discussion and enumeration of the criteria applied to determine whether individual components would be susceptible to IASCC and/or void swelling.

Response – (Rev. 1)

Technical evaluation report (APR1400-Z-M-NR-14017-P/NP, Rev.0, Evaluation of Irradiation Assisted Stress Corrosion Cracking and Void Swelling on Reactor Vessel Internals, December 2014) is added as reference (Reference 13) and will be submitted for detailed information of irradiation effects on reactor vessel internals. The report shows specific criteria including neutron fluence values, stress values, temperatures, and results of the analyses for irradiation assisted stress corrosion cracking and void swelling for APR1400 plant.

Impact on DCD

DCD Tier 2, Sections 4.5.2.6 and 4.5.4 will be revised 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

There is no impact on any Technical, Topical or Environmental Reports.

APR1400 RVI during the 60 year design life. In addition, note that the integrity of the APR1400 RVI is to be also verified through the inservice inspection program which is to be developed based on Section XI requirements of the ASME Code (COL 5.3 (4)).

Add.
More detailed information of irradiation effects on the APR1400 RVI can be found in Reference 13.

4.5.3 Combined License Information

No COL information is required with regard to Section 4.5.

4.5.4 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. 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.

4. ASME Boiler and Pressure Vessel Code, Section II, "Materials," The American Society of Mechanical Engineers, the 2007 Edition with the 2008 Addenda.

5. ASME Boiler and Pressure Vessel Code Section IX, "Welding and Brazing Qualifications," The American Society of Mechanical Engineers, the 2007 Edition with the 2008 Addenda.

6. Regulatory Guide 1.84, "Design, Fabrication and Materials Code Case Acceptability ASME Section III Division 1," Rev. 36, U.S. Nuclear Regulatory Commission, August 2014.

7. Regulatory Guide 1.44, "Control of the Use of Sensitized Stainless Steel," Rev. 1, U.S. Nuclear Regulatory Commission, March 2011.

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

9. Regulatory Guide 1.28, "Quality Assurance Program Criteria (Design and Construction)," Rev. 4, U.S. Nuclear Regulatory Commission, June 2010.

10. ASME NQA-1,"Quality Assurance Requirements for Nuclear Facility Applications," The American Society of Mechanical Engineers, 2008 Edition with 2009 Addenda.

11. Regulatory Guide 1.71, "Welder Qualification for Areas of Limited Accessibility," Rev. 1, U.S. Nuclear Regulatory Commission, March 2007.

12. ANATECH Report, ANA-05-R-0684, Rev. 3.12, "Installation & User's Manual for Version 3.12 of Constitutive Model for Irradiated Austenitic Stainless Steels for Use with ANSYS" April, 2010.

Add.
13. Technical Evaluation Report, APR1400-Z-M-NR-14017, Rev. 0, "Evaluation of Irradiation Assisted Stress Corrosion Cracking and Void Swelling on Reactor Vessel Internals" December, 2014.