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.:	**432-8377**
SRP Section:	**19 – Probabilistic Risk Assessment and Severe Accident Evaluation Section: 19**
Application Section:	**19**
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Question No. 19-62

10 CFR 52.47(a)(23) states that a design certification (DC) application for light-water reactor designs must contain an FSAR that includes a description and analysis of design features for the prevention and mitigation of severe accidents, e.g., challenges to containment integrity caused by core-concrete interaction, steam explosion, high-pressure core melt ejection, hydrogen combustion, and containment bypass.

APR1400 design control document (DCD) Rev. 0, Section 19.2.3.3.5.1.1, states that in-vessel steam explosion analysis is performed to confirm the applicability of the NRC Fuel-Coolant Interactions (FCI) expert review group OECD/NEA FCI specialist conclusions to the APR1400 design. The applicant provided this in-vessel steam explosion analysis in APR1400-E-P-NR-14003-P, "Severe Accident Analysis Report." Rev. 0, Appendix D, "Severe Accident Analysis Report for FCI." Add text to the DCD to describe the in-vessel steam explosion analysis performed including key assumptions, methodology, and key results.

Response – (Rev. 1)

As addressed in Section 3 of APR1400-E-P-NR-14003-P, "Severe Accident Analysis Report", Rev. 0, Appendix D, In-Vessel Steam Explosion (IVSE) analysis has been done to confirm IVSE does not lead to the reactor vessel failure. Methodology for IVSE study is categorized as follows:

- Setup the initial and boundary condition (Section 3.2 in Appendix D)
- Evaluate the enegertic load due to IVSE by using TEXAS-V (Section 3.3 and 3.4 in Appendix D)
- Assess the reactor vessel lower head integrity against the load given by TEXAS-V (Section 3.5 and 3.6 in Appendix D)

To incorporate the requested information in DCD the text including key assumptions, methodology, and key results are added in DCD Section 19.2.3.3.5.1.1 and 19.2.3.3.5.2.1 as shown in the Attachment.

Impact on DCD

DCD Tier 2, Subsection 19.2.3.3.5.1.1 and 19.2.3.3.5.2.1 is 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 Report.

19.2.3.3.5.1 Analysis Methodology

19.2.3.3.5.1.1 In-Vessel Steam Explosion (IVSE)

The alpha-mode failure caused by IVSE has been considered as a threat to containment integrity for many years. The FCI expert review group sponsored by the NRC concluded in NUREG-1116 (Reference 21) and NUREG-1524 (Reference 22) that probability of this failure was vanishingly small or physically unreasonable. The OECD/NEA FCI specialist meeting (Reference 23) confirmed this conclusion. Therefore, the IVSE analysis is performed to confirm the applicability of the experts' conclusions to the APR1400 design.

The analysis consists of determination of initial and boundary conditions for the IVSE including corium and coolant characteristics, evaluation of pressure loads with TEXAS-V (Reference 24), and evaluation of reactor vessel lower head structural integrity against the pressure loads.

19.2.3.3.5.1.2 Ex-Vessel Steam Explosion (EVSE)

EVSE has been considered as one of the important threats to containment integrity for many years although no specific requirements are stated in the CFRs. Therefore, the EVSE analysis aims analytically to confirm the maintainability of containment integrity by employing a mechanistic FCI code to calculate EVSE pressure loads. The APR1400 specific analysis consists of four steps:

a. Selection of the initial and boundary conditions for the base case analysis based on MAAP analysis results

b. Evaluation of pressure loads with TEXAS-V (Reference 24) for the base case analysis

c. Assessment of uncertainties associated with the pressure load evaluation

d. Evaluation of containment structural integrity against the pressure loads

The base case of the EVSE analysis is assumed to be a case where the vessel failed at the bottom center of the RPV due to the in-core instrument guide tube ejection resulting in the ejection of oxidic core debris into a subcooled pool of water in the reactor cavity.

19.2.3.3.5.2 Analysis Result

19.2.3.3.5.2.1 In-Vessel Steam Explosion

The key physical processes that can influence in-vessel steam explosions for PWRs are (a) melt relocation into the lower plenum, (b) corium jet breakup and coarse mixing formation in the lower plenum, (c) triggering of coarse mixing, (d) energetic FCIs, and (e) pressure loads to the upper and lower vessel heads and their responses.

Both NUREG-1116 and NUREG-1524, written by the NRC-sponsored Steam Explosion Review Group, concluded that the potential for alpha-mode failure is vanishingly small or physically unreasonable. The OECD/Committee on the Safety of Nuclear Installations (CSNI) also confirmed the conclusion of NUREG-1524 and concluded that the alpha-mode failure issue was resolved from a risk perspective.

Because the APR1400 design is not significantly different from current PWRs, the NUREG-1524 conclusions are applicable to the APR1400 design, thus no mitigation features are provided to prevent or mitigate IVSE.

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19.2.3.3.5.2.2 Ex-Vessel Steam Explosion

The initial and boundary conditions for EVSE are largely dependent upon the in-vessel severe accident progression, severe accident management procedure, and vessel failure modes. Thirteen severe accident sequences were chosen to cover the spectrum of key variable parameters and thus characterize the initial and boundary conditions for EVSE analysis. The key parameters considered include corium discharge rates, corium thermal conditions, cavity conditions, and related parameters.

The result of analysis using the MAAP code provided the initial conditions for the TEXAS-V code. TEXAS-V was then used to calculate the peak pressure due to EVSE. The pressure at the nearest cavity wall was then estimated by the TNT method (Reference 25).

The reactor cavity and RPV column support have to maintain structural integrity in events such as an ex-vessel steam explosion. The reactor cavity and RPV column support is

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Initial and boundary conditions for the IVSE are constructed under the conservative assumptions such as the jet falling at the center of the vessel and the ignoring of the complex internal structure disruption. The key parameters considered include corium temperature, jet diameter, jet velocity, water level, and water temperature. To reflect the uncertainty pertinent to the relocation, multi-jet configuration is also examined. TEXAS-V was then employed to evaluate the peak pressure due to IVSE.

The stress analysis for the reactor vessel lower head against the dynamic loads given by IVSE event was performed with the conservative failure criteria of the pressure vessel. ~~The stress analysis indicates that no thread of the APR1400 lower head due to the IVSE is expected.~~

ABAQUS 6.10 (Reference 43) is used for the stress analysis FEM tool. Conservative design criteria proposed by Shockey (Reference 44) who adopted the concept of ductile fracture apparatus based on void generation and growth is used. According to Shockey suggestion, a maximum plastic equivalent strain of 11% is used in failure criteria. The stress analysis indicates that no threat of the APR1400 lower head due to the IVSE is expected.

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38. APR1400-K-X-ER-14001-NP, "APR1400 Design Certification: Applicant's Environmental Report - Standard Design Certification," KHNP, December 2014.

39. NEI 05-01, "Severe Accident Mitigation Alternatives (SAMA) Analysis - Guidance Document," Rev. A, Nuclear Energy Institute (NEI), November 2005.40. Bureau of Labor Statistics' Producer Price Index for the commodity of "Electric Power" (BLS 2011| Producer Price Index-Commodities: Series Id: WPU054| 2012/1993) (retrieved March 21, 2013).

41. NUREG/BR-0184, "Regulatory Analysis Technical Evaluation Handbook," U.S. Nuclear Regulatory Commission, 1997.

42. APR1400-E-P-NR-14006-P, "Severe Accident Mitigation Design Alternatives for the APR1400," Rev. 0, KHNP, December 2014.

43. Abaqus Analysis User's Manual, vol 2-Abaqus 6.10

44. Shockey, D. A et al., "Kinetics of void development in fracturing A533B tensile bars," J. Pressure Vessel Technology, 102(1980), pp 14-21