KHNPDCDRAIsPEm Resource

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Sent: Friday, April 22, 2016 9:42 AM
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Cc: Wagage, Hanry; Mrowca, Lynn; Steckel, James; Williams, Donna
Subject: APR1400 Design Certification Application RAI 467-8394 (19 - Probabilistic Risk Assessment and Severe Accident Evaluation)
Attachments: APR1400 DC RAI 467 SPRA 8394.pdf

KHNP,

The attachment contains the subject request for additional information (RAI). This RAI was sent to you in draft form. Your licensing review schedule assumes technically correct and complete responses within 30 days of receipt of RAIs.

Please submit your RAI response to the NRC Document Control Desk.

Thank you,

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United States Nuclear Regulatory Commission
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REQUEST FOR ADDITIONAL INFORMATION 467-8394

Issue Date: 04/22/2016
Application Title: APR1400 Design Certification Review – 52-046
Operating Company: Korea Hydro & Nuclear Power Co. Ltd.
Docket No. 52-046
Review Section: 19 - Probabilistic Risk Assessment and Severe Accident Evaluation
Application Section:

QUESTIONS

19-102

10 CFR 52.47(a)(23) states that a 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. Revise the DCD to address the following.

A. APR1400 DCD Rev. 0 Section 19.2.3.2.1 states that the in-vessel core melt progression contains considerable uncertainty relating to the following:
   a. Potential for in-vessel steam explosion
   b. Interaction between core debris and internal vessel structures
   c. Time and mode of vessel failure
   d. Composition of the core debris released at vessel failure
   e. Amount of in-vessel hydrogen generation
   f. In-vessel fission-product release and transport
   g. Retention of fission products and other core materials in the RCS

   The DCD needs to describe how uncertainties relating to the above items were addressed.

B. APR1400 DCD Rev. 0 Section 19.2.3.2.2 Ex-Vessel Melt Progression does not list or describe uncertainty relating to the ex-vessel core melt progression. List and explain how uncertainty relating to the ex-vessel core melt progression was addressed.

19-103

SRM on SECY-93-087 states the following: The Commission approves the staff's recommendation that the applicant for design certification for a passive or evolutionary PWR assess design features to mitigate the amount of containment bypass leakage that could result from steam generator tube ruptures.

SECY-93-087 states the following:

> The staff concludes that containment bypass resulting from SGTRs can be a significant challenge to containment integrity. Therefore, the staff concludes that the plant designer should consider design features that would reduce or eliminate containment bypass leakage in such a scenario. The following features could mitigate the releases associated with a tube rupture:

- a highly reliable (closed loop) steam generator shell-side heat removal system that relies on natural circulation and stored water sources;
- a system which returns some of the discharge from the steam generator relief valve back to the primary containment; or,
- increased pressure capacity on the steam generator shell side with a corresponding increase in the safety valve setpoints.

Describe APR1400 design features that could mitigate the releases associated with a tube rupture and update the DCD as necessary.

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