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.:	**433-8363**
SRP Section:	**SRP 19**
Application Section:	**19.1**
Date of RAI Issue:	**03/08/2016**

Question No. 19-83

All design certification applications under 10 CFR 52.47 shall satisfy the requirements of 10 CFR 52.47(a)(27) that a description and results of a design-specific PRA be provided as part of the application for design certification. The estimate of containment fragility as a function of pressure and associated temperature loads needs to be developed for use in the design-specific PRA.

APR1400 DCD Tier 2, Figure 19.1-47: Total Containment Fragility Curve shows the results of the containment fragility analysis, which includes the sum of the probability density functions for each failure mode considered. Section 19.1.4.1.2.2 and Table 19.1-28 provide the failure locations and failure modes considered for the containment ultimate pressure capacity and also for the containment fragility calculation.

a. In Section 19.1.4, the applicant lacks a description of a specific analytical process used to determine the containment pressure fragility and associated failure modes. The staff reviewed the Ultimate Pressure Capacity Report (APR1400-K-P-NR-013605-P) and requests the applicant provide the following information in the DCD:

    i An explanation of whether severe accident temperature effects on material properties were considered.

    ii A clarification of the basis for the assumed temperature (400 F) and the relationship between that temperature and the temperature reduction factors in Section 5.6 of the report.

    iii A description of the consideration of uncertainties

    iv A description of how the uncertainty for the different parameters considered in the analysis is estimated and aggregated.

b. In Section 19.1.4.2.1.2.2, the applicant lists the personnel emergency exit airlock as a failure location contributing to the containment pressure fragility. There are no corresponding failure modes included in Table 19.1-28 or Figure 19.1-47. Describe the basis for the selection of failure locations and failure modes considered and confirm whether the failure of the personnel emergency exit airlock was included in the analysis.

Response - (Rev. 1)

a. Containment Ultimate Pressure Capacity Analysis (UPC) in DCD Section 19.1.4.2.1.2.2 is revised as shown in Attachment 1.

   i The bounding harsh environmental conditions are determined by simulating a broad spectrum of accident sequences including LBLOCA, MBLOCA, SBLOCA, TLOFW, and SBO, and dominant PRA sequences using the MAAP code. Long term atmospheric temperature after temperature spikes is enveloped by a temperature of 460 K (368 °F) (Reference 19.2.3.3.7.2.1). Based on long term temperature, 400°F is conservatively assumed for the strength properties of reinforcing and prestressing steels.

   ii The relationship between the long term temperature 400°F and the temperature reduction factor is clarified based on The Structural Engineer, “The effect of elevated temperatures on the strength properties of reinforcing and prestressing steels”, M. Holmes, R.D. Anchor, G.M.E.Cook & R.N.Cook, Vol. 60B, No. 1, March 1982. This report shows that the typical strength properties of reinforcing and prestressing steels tested at room temperature after heating to an elevated temperature and gradually reduced under the conditions of concrete cracking and reinforcement yielding.

   iii For controlling failure modes, the uncertainties associated with the median capacities are evaluated. The uncertainties, βM and βS, represent variability due to a lack of knowledge related to difference between the analytical model and the real structure. The modeling uncertainty (βM) such as internal force distribution, failure criteria, the use of empirical formulae in tendon and reinforcing steel placement, and variability in force-deformation relations is considered. The strength uncertainties (βS) are associated with variability related to the material resistance. The variability in concrete strength, tendon and reinforcing steel strength, steel liner strength, and the influence of elevated temperature on the material strengths are considered.

   iv The random variable (uncertainty in the median pressure capacity) is assumed to be lognormally distributed with unity median and lognormal standard deviation (βU). This assumption is considered to be appropriate, since parameters that influence containment capacity, such as material strengths, are best represented by lognormal distributions. The uncertainty (βU) can be developed from combinations such as SRSS of the strength uncertainties (βS) and modeling uncertainty (βM).

b. The selection of emergency exit airlock failure location was based on the review of research results such as “Containment Integrity Research at Sandia National Laboratories”, NUREG/CR-6906 and design UPC outputs (structural drawings, containment penetration detail). These results show that the unhomogeneous part of containment is relatively vulnerable than other parts and the failure modes are listed based on such results.

The personnel emergency exit airlock was included in the analysis and Table 19.1-28 of DCD Section 19.1 is replaced as shown in Attachment 2.

The personnel airlock and personnel emergency exit airlock have the same dimension and material properties, therefore both airlocks have the identical pressure capacity.

Impact on DCD

Table 19.1.4.2.1.2.2 is revised as shown in Attachment 1 and Table 19.1-28 is replaced as shown in Attachment 2.

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 Environment Report.

This issue is of primary concern to BWR plants because of the drywell design. For completeness of the PRA, this mode of failure was considered in the APR1400 PRA even though the pathways for debris transport out of the reactor cavity are to interior containment building compartments away from the containment wall. The probability of this failure mode was assigned a negligible value.

i. Failure of Containment Building Penetrations

Failure of containment building penetrations (electrical, fluid, equipment hatch, personnel hatch, etc.) was explicitly evaluated in the analysis of the containment overpressure capacity and found to be significantly less important than overpressure failure of the cylinder wall. Temperature-induced penetration failures were treated in the APR1400 PRA.

To model containment responses for most accident sequences, a general CET is developed. Special CETs were developed for the containment bypass, containment isolation failure, and containment failure before RV breach. These CETs properly considered all pertinent containment failure modes identified for the APR1400 containment. The important phenomena that can affect the containment failure modes and the source terms are also addressed in the CETs. The questions and important events that are used in significant references (e.g., NUREG-1150 and previous Level 2 PRAs for other plants) are reviewed and included in the APR1400 PRA CETs.

The containment event trees are shown in Figure 19.1-42 through Figure 19.1-46.

19.1.4.2.1.2.2 Containment Ultimate Pressure Capacity Analysis

In order to evaluate the likelihood of containment failure for various accident progression phenomena, it is necessary to determine a realistic pressure at which the containment would fail. In nuclear power plants, the containment design failure pressure is 2 to 3 times less than the realistic, as-built failure pressure. Therefore, a best-estimate assessment of the APR1400 containment was performed. This section summarizes the evaluation and results.

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~~A plant-specific containment structural analysis was performed to determine the ultimate pressure capacity of the APR1400 containment building, and to identify the failure modes. Potential modes considered include:~~

Insert A on next page

a. Membrane failure

b. Cylindrical wall at the basemat

c. Failure of the basemat

d. Failure of the equipment hatch

e. Failure of the personnel access airlock

f. Failure of the personnel emergency exit airlock

g. Failure of the fuel transfer tube

Insert B on next page

Replace with C

~~In this analysis, the several failure modes that are identified in the containment structural analysis can be classified by their failure sizes into two groups, which are defined in NUREG-1150 and NUREG/CR-6906 (Reference 35).~~

~~Insert B on next page~~

a. A leak is defined as a containment breach that would arrest a gradual pressure buildup, but would not result in containment depressurization in less than 2 hours. The typical leak size is evaluated to be on the order of 9.29×10^{-3} m^2 (0.1 ft^2).

b. A rupture is defined as a containment breach that would arrest a gradual pressure buildup and would depressurize the containment within 2 hours. The typical rupture size is evaluated to be on the order of approximately 9.29×10^{-2} m^2 (1.0 ft^2).

The failure modes and their results are as presented in Table 19.1-28.

A probability density function was calculated for each potential failure mode, and summed together to estimate a total fragility curve. The results of this analysis are presented in Figure 19.1-47. These results were used in the Level 2 phenomenological evaluations for leak and rupture failure pressures of the containment.

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A

A plant-specific containment structural analysis was performed to determine the ultimate pressure capacity (UPC) of the APR1400 containment building, and to identify the failure modes. The selection of failure modes was based on the review of research results from NUREG/CR-6906 (Reference 35) and design UPC outputs (structural drawings, containment penetration detail). These results show that the unhomogeneous part of containment is relatively vulnerable than other parts and the failure modes are listed based on such results. Potential modes considered include:

B

The bounding harsh environmental conditions are determined by simulating a broad spectrum of accident sequences including LBLOCA, MBLOCA, SBLOCA, TLOFW, and SBO, and dominant PRA sequences using the MAAP code. Long term atmospheric temperature after temperature spikes is enveloped by a temperature of 460 K (368 °F) (Reference 19.2.3.3.7.2.1). Based on long term temperature, 400°F is conservatively assumed for the strength properties of reinforcing and prestressing steels. The relationship between the long term temperature 400°F and the temperature reduction factor is clarified based on reference 59. This report shows that the typical strength properties of reinforcing and prestressing steels tested at room temperature after heating to an elevated temperature and gradually reduced under the conditions of concrete cracking and reinforcement yielding.

For controlling failure modes, the uncertainties associated with the median capacities are evaluated. The uncertainties, β_M and β_S, represent variability due to a lack of knowledge related to difference between the analytical model and the real structure. The modeling uncertainty (β_M) such as internal force distribution, failure criteria, the use of empirical formulae in tendon and reinforcing steel placement, and variability in force-deformation relations is considered. The strength uncertainties (β_S) are associated with variability related to the material resistance. The variability in concrete strength, tendon and reinforcing steel strength, steel liner strength, and the influence of elevated temperature on the material strengths are considered. The random variable (uncertainty in the median pressure capacity) is assumed to be lognormally distributed with unity median and lognormal standard deviation (β_U). This assumption is considered to be appropriate, since parameters that influence containment capacity, such as material strengths, are best represented by lognormal distributions. The uncertainty (β_U) can be developed from combinations such as SRSS of the strength uncertainties (β_S) and modeling uncertainty (β_M).

C

In this analysis, the several failure modes that are identified in the containment structural analysis can be classified by their failure sizes into two groups, which are defined in NUREG-1150 and NUREG/CR-6906 (Reference 35).

42. NUREG-1921, "EPRI/NRC-RES Fire Human Reliability Analysis Guidelines," U.S. Nuclear Regulatory Commission, November 2009.

43. EPRI 1016735, "Fire PRA Methods Enhancements: Additions, Clarifications, and Refinements to EPRI 1019189," Electric Power Research Institute, December 2008.

44. NUREG/CR-4527, "An Experimental Investigation of Internally Ignited Fires in Nuclear Power Plant Control Cabinets, Part II: Room Effects Tests," U.S. Nuclear Regulatory Commission, April 1987.

45. Regulatory Guide 1.102, "Flood Protection for Nuclear Power Plants," U.S. Nuclear Regulatory Commission, September 1976.

46. EPRI 1021086, "Pipe Rupture Frequencies for Internal Flooding Probabilistic Risk Assessments (PRAs)," Electric Power Research Institute, October 2010.

47. NUREG/CR-6144 (BNL-NUREG-52399), "Evaluation of Potential Severe Accidents During Low Power and Shutdown Operations at Surry, Unit 1," U.S. Nuclear Regulatory Commission, June 1994.

48. Inspection Manual Chapter 0609, Appendix G, "Shutdown Operations Significance Determination Process," U.S. Nuclear Regulatory Commission, February 2005.

49. NUMARC 93-01, "Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," Nuclear Energy Institute, July 2000.

50. NEI 00-04, "10 CFR 50.69 SSC Categorization Guideline," Rev. 0, Nuclear Energy Institute, July 2005.

51. CAFTA 6.0b, Software Manual, EPRI, Palo Alto, CA, 2014.

52. NUREG/CR-7114, "A Framework for Low Power/Shutdown Fire PRA," U.S. Nuclear Regulatory Commission, September 2013.

53. NUREG/CR-7150, "Joint Assessment of Cable Damage and Quantification of Effects from Fire (JACQUE-FIRE)," May 2014.

54. The Structural Engineer, "The effect of elevated temperatures on the strength properties of reinforcing and prestressing steels", M. Holmes, R.D. Anchor, G.M.E.Cook & R.N.Cook, Vol. 60B, No. 1, March 1982.

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Table 19.1-28

Containment Failure Modes and Results

Failure Mode		Force	Median Pressure (psi)	Logarithmic Standard Deviation	Failure Size Category
Cylinder Wall	Hoop Direction	Membrane	204.79	0.20	Rupture
	Meridional Dire.	Membrane	257.66	0.19	Rupture
Dome	Hoop Dire. (above 45 deg.)	Membrane	221.90	0.23	Rupture
	Meridional Dire. (under 45deg.)	Membrane	237.31	0.18	Rupture
Basemat -Cylinder Wall Junction	Radial Dire.	Shear	214.80	0.29	Rupture
	Meridional Dire.	Moment	224.28	0.16	Rupture
Basemat	Screen out	-	-	-	Rupture
Equipment Hatch	Spherical Hatch Cover	Buckling	225.71	0.15	Rupture
	Welded Stud	Shear	251.69	0.11	Rupture
	Wall-Hatch Junction	Shear	459.10	0.29	Rupture
Personnel Air lock (Personnel Emergency Exit Air lock)	Spherical Hatch Cover (Bulkhead plate)	Buckling	222.18	0.15	Rupture
	Welded Stud	Shear	253.03	0.11	Rupture
	Wall-Hatch Junction	Shear	443.92	0.24	Rupture
Fuel Transfer Tube	Blind Flange	-	223.08	0.15	Rupture
	Sleeve	-	281.03	0.15	Rupture
Fuel Transfer Tube	Welded Stud	Shear	420.55	0.15	Rupture
	Wall-Hatch Junction	Shear	752.43	0.24	Rupture
Liner Tearing of Equipment Hatch	-	-	188.0	0.15	Leak

~~Personal emergency exit air lock~~	~~Spherical Hatch Cover (Bulkhead plate)~~	~~Buckling~~	~~222.18~~	~~0.15~~	~~Rupture~~
	~~Welded Stud~~	~~Shear~~	~~253.03~~	~~0.11~~	~~Rupture~~
	~~Wall-Hatch Junction~~	~~Shear~~	~~443.92~~	~~0.24~~	~~Rupture~~