### Regulatory Issues Regarding the Ultimate Pressure Capacity Assessment of the Containment Buildings and Relevant Research Results

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#### 1. Introduction

Following the Three Mile Island accident in 1979 and the Fukushima accident in 2011, new safety standards for nuclear power plants were established to ensure their ongoing safety beyond design-based accidents [1]. Containment buildings are the primary shielding structures in nuclear power plants, and their integrity is critical to nuclear safety [2]. The purpose of this study is to summarize regulatory issues regarding the deterministic and probabilistic ultimate pressure capacity assessments of the containment buildings. The relevant research results [3] are reviewed for each regulatory issue.

# 2. Deterministic ultimate pressure capacity assessment

#### 2.1 Failure criteria

RG 1.216 [4] specifies the free-field for determining functional failure in liner as a reasonable distance away from discontinuities, however each user may change the allowable free-field locations. The internal pressure capacity of containment buildings at a functional failure may be quantitatively adjusted by shifting the free-field position. Thus, there is an argument to be made for evaluating functional failure in liners.

The ultimate pressure capacity of the containment buildings was evaluated using finite element analysis in several previous works [5,6,7,8]. Choun and Park [5] and Alhanaee et al. [6] used a 0.4 % principal strain of the liner at the midheight of the wall as a leak failure criteria, rather than the 0.4 % global free-field strain of liners and rebars specified in RG 1.216. Basha et al. [7], on the other hand, assumed that the stress concentration area adjacent to the equipment hatch was the most vulnerable to leaks and used a strain of 6.4 % in liners as a failure criteria. Choun and Park [5] used the 0.8 % total average tensile strain of the tendons as a rupture failure criteria in finite element analysis proposed in RG 1.216. Additionally, some studies [5,8] reported rupture failure using a 1% tensile strain of tendons, which is considerably greater than the RG 1.216's failure criteria.

#### 2.2 Penetration model analysis

RG 1.216 requires an evaluation to demonstrate that leaks in penetrations, bolted joints, seals, doorways, and bellows are small enough until their pressures reach the ultimate pressure capacity [4]. If the leaks are large, the ultimate pressure capacity was determined as the specific pressures at which the allowable leaks occur. However, few analyses of the penetration model have been carried out so far. Basha et al. [7] evaluated the functional failure of the penetration model by incorporating a modification factor such as the triaxiality factor. SNL conducted penetration modeling using the actual drawing and carried out a series of numerical analyses to predict the liner strain in concentrated areas of the penetration model [9]. SNL also proposed a method for applying a reduction factor to uniaxial fracture strains of the steels and a method for applying an amplification factor to strains of the liners obtained from global behavior analyses [10].

## 3. Probabilistic ultimate pressure capacity assessment

#### 3.1 Analysis methodology

In regulatory field, deterministic analysis has been performed using three-dimensional finite element models, while probabilistic analysis has been performed using an analytical calculation method based on conservative assumptions. As a result, different ultimate pressure capacity values for the same containment building may exist, making it difficult to establish a relationship between them. To improve the precision of the analysis and to use a more technically sophisticated method, 3D finite element analysis can be recommended to evaluate the probabilistic ultimate pressure capacity. Previous studies [11,12,13,14] used the finite element analysis method to evaluate the probabilistic ultimate pressure capacity of the containment buildings.

#### 3.2 Failure criteria

RG 1.216 is the regulatory guideline that only applies to deterministic analyses of ultimate pressure capacity and does not apply to probabilistic evaluations of ultimate pressure capacity. However, the conservative analysis technique, material model, and failure criteria stated in RG 1.216 seem to be applicable for determining the probabilistic ultimate pressure capacity. As a result of previous studies [12,13,14,15], various strain limits for concrete, liner, reinforcing bar, and tendon were used as failure criteria for each constituent material model to determine the probabilistic ultimate pressure capacity. Hahm et al. [12], for example, evaluated the probabilistic ultimate pressure capacity of a PWR-type containment building using liner and tendon failure criteria in SRP 3.8.1 (liner strain of 0.8 % and tendon strain of 0.8 %) [15]. Using the liner failure criteria in RG 1.216, Jin et al. [14] computed the probabilistic ultimate pressure capacity (liner strain of 0.4 %).

#### 3.3. Uncertainty of materials and sampling methods

The results of multiple structural failure types at the wall, dome, and wall-foundation joint are currently combined to produce a fracture probability for rupture, whereas leaks are expected to be most prone to failure by tearing of the liner plate around the equipment hatch. However, the probability of failure needs to be compared not only at the equipment hatch, which is prone to leaks, but also at the upper part of the dome and the midheight of the wall for probabilistic ultimate pressure capacity. Previous studies [11,13] used the Monte Carlo Sampling (MCS) method for random sampling of the uncertainty factors corresponding to material and structural characteristics. The Latin Hyper-Cube Sampling (LHCS) method was also utilized to ensure high reliability results even with a small number of samples [12,14].

#### 3.4 Penetration model analysis

For probabilistic assessment of the penetration model, a weak point is generally assumed to be an area of discontinuous thickness between the concrete and the liner plate. As a result, a separate and detailed analysis of the penetration model is required to assess the leakage paths caused by deformation of the bolt connection material and the sealing material of penetration parts, as well as to estimate the structural weakness of penetration parts. The NUREG/CR-6809 [9] literature can be referred to determine the fragility of the penetration parts.

#### 4. Conclusions

The purpose of this study is to summarize regulatory issues regarding the assessment of the ultimate pressure capacity of the containment buildings and to discuss the relevant research results related to deterministic and probabilistic evaluations. The results of the study will be useful to enhance understanding of regulatory issues on ultimate pressure capacity.

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#### REFERENCES

[1] IAEA, Safety of Nuclear Power Plants: Design, IAEA No. SSR-2/1, IAEA, Vienna, 2012.

[2] Tong, L., Zhou, X., Cao, X., Ultimate pressure bearing capacity analysis for the prestressed concrete containment. Annals of Nuclear Energy, 121, 582-593, 2018.

[3] NSTAR-21NS21-215, Research status on assessment of ultimate pressure capacity of containment building, 2021.

[4] Regulatory Guide 1.216, Containment structural integrity evaluation for internal pressure loadings above design-basis pressure, Rev.0, U.S. Nuclear Regulatory Commission, 2010.

[5] Choun, Y.S. and Park, H.K., Containment performance evaluation of prestressed concrete containment vessels with fiber reinforcement. Nuclear Engineering and Technology, 47(7), 884-894, 2015.

[6] Alhanaee, S., Yi, Y.S. and Schiffer, A., Ultimate pressure capacity of nuclear reactor containment buildings under unaged and aged conditions. Nuclear Engineering and Design, 335, 128-139, 2018.

[7] Basha, S.M., Singh, R.K., Patnaik, R., Ramanujam, S., Kushwaha, H.S., and Venkat R.V., Predictions of ultimate load capacity for pre-stressed concrete containment vessel model with BARC finite element code ULCA. Annals of Nuclear Energy, 30(4), 437-471, 2003.

[8] Lee, S.K., Song, Y.C., Han, S.H. and Kwon, Y.G., Evaluation of ultimate pressure capacity of light water reactor containment considering aging of materials. Journal of the Korea Institute for Structural Maintenance and Inspection, 5, 147-154, 2001.

[9] NUREG/CR-6809, Posttest Analysis of the NUPEC/NRC
1:4 Scale Prestressed Concrete Containment Vessel Model, ANATECH Corporation & Sandia National Laboratories, 2003.
[10] NUREG/CR-6706, Capacity of Steel and Concrete Containment Vessels with Corrosion Damage, Sandia National Laboratories, 2000.

[11] Kralik, J., Probabilistic safety analysis of the nuclear power plants in lovakia. Journal of KONBiN, 2,3, 35-48, 2010. [12] Hahm, D.K., Park, H.K. and Choi, I.K., Assessment of the internal pressure fragility of the PWR containment building using a nonlinear finite element analysis. Journal of the Computational Structural Engineering Institute of Korea, 27(2), pp. 103-111, 2014.

[13] Zhou, L., Li, J.B., Zhong, H., Lin, G. and Li, Z.C., Fragility comparison analysis of CPR1000 PWR containment subjected to internal pressure. Nuclear Engineering and Design, 330, 250-264, 2018.

[14] Jin, S., Li, Z.C., Lan, T.Y. and Gong, J.X., Fragility analysis of prestressed concrete containment under severe accident condition. Annals of Nuclear Energy, 131, 242-256, 2019.

[15] Standard Review Plan 3.8.1, Concrete Containment, Rev. 4, US NRC, (NUREG-0800), 2013.