Failure Mode Estimation of Wolsong Unit 1 Containment Building with respect to Severe Accident Condition

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1. Introduction

The containment buildings in a nuclear power plant (NPP) are final barriers against the exposure of harmful radiation materials at severe accident condition. Since the accident at Three Mile Island nuclear plant in 1979, it has become necessary to evaluate the internal pressure capacity of the containment buildings for the assessment of the safety of nuclear power plants [1-3]. According to this necessity, many researchers including Yonezawa et al. [4] and Hu & Lin [5] analyzed the ultimate capacity of prestressed concrete containments subjected to internal pressure which can be occurred at sever accident condition. Especially in Wolsong nuclear power plant, the Unit 1 containment structures were constructed in the late 1970 to early 1980, so that the end of its service life will be reached in near future. Since that the complete decommission and reconstruction of the NPP may cause a huge expenses, an extension of the service time can be a cost-effective alternative. To extend the service time of NPP, an overall safety evaluation of the containment building under severe accident condition should be performed.

In this study, we assessed the pressure capacity of Wolsong Unit 1 containment building under severe accident, and estimated the responses at all of the probable critical areas. Based on those results, we found the significant failure modes of Wolsong Unit 1 containment building with respect to the severe accident condition.

On the other hand, for the aged NPP, the degradation of their structural performance must also be explained in the procedure of the internal pressure capacity evaluation. Therefore, in this study, we performed a parametric study on the degradation effects and evaluated the internal pressure capacity of Wolsong Unit 1 containment building with considering aging and degradation effects.

2. Methods and Results

In this section some of the techniques used to model the loads and structural systems are described.

2.1 Load Models

In this study, we modeled the temperature and internal pressure loads at the severe accident. Some mechanical properties of concrete such as compressive strength and modulus elasticity vary significantly with the ambient temperature [6]. Compressive strength decreases rapidly with increasing temperature except in the range of 100~400 °C. The Young's modulus decreases almost linearly as temperature increases. High temperature effect on the containment performance was studied by some researchers [5, 7]. Hence, the ambient temperature in severe accident condition should be considered for the complete estimation of the safety of containment buildings.

Choi & Park [8] evaluated the temperature and pressure behaviors in the containment building according to the accident scenario of Wolsong Unit 1 NPP. For the selected accident sequences of the NPP, they performed the thermal hydraulic analysis to evaluate the pressure and temperature condition in the containment building. The simulated pressure and temperature condition for the simulated accident scenario are shown in Figure 1. The internal pressure is increased up to 1,000 kPa and the temperature is up to 440°K(167°C) under the most severe accident condition.



Fig. 1. Pressure and temperature history in atmosphere of upper dome under the simulated accident sequences

2.2 Structural Models

For the overall re-assessment of structural safety and fragility of the Wolsong Unit 1 containment building, Korea Institute of Nuclear Safety (KINS) suggested following seven probable failure modes under severe accident condition.

- Flexural or shear failure at the bottom of a perimeter wall
- Flexural or shear failure at the base slab
- Failure at the equipment & personal airlocks
- Failure of nuclear fuel transfer tubes
- Failure at the pipe penetration area of the containment building
- An interaction between the containment building and the service buildings

• Failure caused by excessive stress of perimeter wall and upper dome

For the consideration of complete failure modes, we modeled the containment building as 3D FE models including the temporary opening and the equipment & personal airlocks (Fig. 2). The FE-based generalpurpose structural analysis program, ABAQUS [9], was adopted as an analysis tool. The reinforcement bars and tendons were modeled using embedded surface and truss elements, respectively. The material nonlinearity of the concrete was implemented by introducing the concrete damaged plasticity model [9]. The tri-linear plasticity model and piecewise linear stress-strain model were used for the material nonlinearity of steel rebars and tendons, respectively. For the modeling of containment system, basically we used the geometric data of the Wolsong Unit 2~4 since that the design document of Unit 1 was very rare and it was difficult to Major differences on the geometric collect. configuration and important material properties between the Unit 1 and 2~4 are summarized in Table 1.

Table 1. Geometric configuration and material properties of Wolsong Units containment buildings

Properties	Unit 1	Unit 2~4
Design temperature (°C)	15~21	135 (Accident)
Radius of perimeter wall (m)	41.46	41.50
Thickness of base slab (m)	1.68	1.50
Young's modulus (MPa)	199,800	193,000
Ultimate tensile strength (MPa)	1,757	1,860
Radius of wire (in)	0.276	0.500



(a) Whole model

(b) Tendon model



2.3 Analysis Results

The ultimate pressure capacity of Wolsong Unit 1 containment building under severe accident was assessed, and the responses at all of the probable critical areas are also estimated. Displacement responses with respect to the internal pressure loading at the center of the perimeter wall and the top of the dome are depicted in Fig. 3. The displacement response at the top of the dome is quite larger than that of the perimeter wall even though those at the ultimate pressure capacity are almost same. Hence, in this figure, we can conclude that the

excessive displacement at the top of dome is quite significant compared to that of perimeter wall. More detailed results for the many parametric variables will be presented in the KNS autumn meeting.



Fig. 3. Displacement responses with respect to internal pressure load at critical points

3. Conclusions

The pressure capacity of Wolsong Unit 1 containment building under severe accident was assessed, and the responses at all of the probable critical areas are estimated. Based on those results, we found the significant failure modes of Wolsong Unit 1 containment building with respect to the severe accident condition.

ACKNOWLEDGEMENT

This research was supported by the Nuclear Research & Development Program of the Ministry of Knowledge Economy, Korea.

REFERENCES

[1] US Nuclear Regulatory Commission, Office of nuclear reactor regulation. Standard review plan of the safety analysis reports for nuclear plants, section 3.8.1, NUREG-0800; 1987.

[2] Amin M, Eberhardt AC, Erler BA. Design considerations for concrete containments under severe accident loads. Nucl Eng Des 1993;145:331-8.

[3] Boeck BD. A review of containment accidents. Nucl Eng Des 1993;145:279-88.

[4] Yonezawa K, Imoto K, Watanabe Y, Akimoto M. Ultimate capacity analysis of 1/4 PCCV model subjected to internal pressure. Nucl Eng Des 2002;212:357-379.

[5] Hu HT, Lin YH. Ultimate analysis of PWR prestressed concrete containment subjected to internal pressure. Int J Pres Ves Pip 2006;83:161-167.

[6] Freskakis GN, Burrow RC, and Debbas EB. Strength properties of concrete at elevated temperature, CONF-7904083, 1972.

[7] Pfeiffer PA and Kennedy JM. Thermal effects in the overpressurization response of reinforced concrete containment. Nucl Eng Des, 1990;120.

[8] Choi I-K and Park S-Y. Temperature load assessment of CANDU containment building at severe accident condition. Tranaction of the Korean Nuclear Society Autumn Meeting, Gyeongju, Korea, Oct. 29-30. 2009.

[9] ABAQUS/Standard 6.8 – User's Manual, 2008, Hibbitt, Karlsson & Sorensen, Inc.