Research of Containment Pressure and Temperature Response

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Abstract. In the paper, the pressure and temperature response of the containment with free volume 1400 m³ and 2000 m³ is analyzed based on the mass and energy release for postulated loss of coolant accident (LOCA). And the relation between containment free volume and containment pressure also is analyzed. The results of calculation have certain guiding significance for the determination of the pressurized water reactor (PWR) volume.

1. Introduction

The reactor, its cooling loops and equipment are enclosed in a containment. As the third barriers of nuclear power generation, it is a key facility to protect the public from over-dose radiation exposure. And it is absolutely vital for assuring the integrity and tightness of the containment.

In PWR nuclear plant, design basis accidents (DBAs) include instantaneous double-end fracture accident of reactor cooling agent system pipeline in containment and the secondary circuit system pipeline. The high temperature and pressure steam-water mixtures sprayed into the containment to cause the containment pressure rising quickly to a peak in a few seconds after the accident. With the security system working, the energy was taken out to the containment, then the pressure and temperature dropped slowly. The thermal hydraulic phenomena occurring in the containment, such as heat transfer between spray water and steam, condensation of water vapor, evaporation of liquid water, heat transfer between meteorological and thermal components and so on.

For a dry containment, its structure and system design should meet the design reference that the containment should be subjected to pressure not exceeding the containment designed pressure after DBAs. There were lots of programs to analyze the containment temperature and pressure. For example, GAO Yingxian et al. [1] analyzed the containment response by the codes of PAREO and MELCOR based on the same mass and energy release for postulated loss of coolant accident, and the comparison of key physical models of these two codes were studied for analyzing the rationality and penalizing of the results; Wang Guodong et al. [2] developed a simplified model to simulate the AP1000 containment pressurization under design basis accidents (DBAs), and the containment response has been performed by WGOTHIC code for comparison purpose. Other business software, such as CONTAIN [3], MELCOR [4], COMMIIX [5] and PCCSAC [6] and so on, also were used to analyze the heat and mass transfer of the containment system. In the paper, a codes was adopted to analyze the relation between containment free volume and containment pressure.
2. Input Data

2.1. Mass and Energy Data of the Breach
The double-end fracture of the cold section is selected as set DBA. When the accident occurs, the inlet and outlet pipelines of the steam generator are closed. The high temperature and pressure mass and energy of the primary circuit released into containment, assuming no heat conduction containment structures.

The optimum estimation procedure RELAP5 was adopted for analysis. After the breach occurred, the primary system was sprayed to the containment until the end of the spraying after the pressure was balanced. Assuming the containment was adiabatic, there was condensation in the containment, so it was not necessary to consider the applicability of condensing relations. The rapid flash of the releasing high temperature and pressure coolant into steam caused the containment pressure keeping rising until the end of spraying. The change of mass and flow rate were as shown in the Figure 1 when the spraying was end after 30 seconds.

![Figure 1. The Change of Mass and Flow Rate in Containment.](image)

2.2. Procedure Introduction
There are lots of containment pressure codes to analyze the containment pressure based on the mass and energy release for postulated loss of coolant accident, such as CONTEMPT-LT, PAREO. This type of procedure uses a lumped parameter approach, which is based on the deviation due to simplification of the momentum equation of experience in nuclear industry and lumped parameter approach limitations determined by extensive testing, and uses conservation analysis model to assure the containment pressure peak.

The program can compute temperature, pressure, mass, energy and the temperature analysis of thermal conductors in the containment, and time-dependent energy transfer between the standby compartments. The program can be used to analyze and calculate the pressured water reactor dry containment, negative containment, and double containment, as well as the standard boiling water reactor suppression system. The paper used the program to model the pressured water reactor dry containment, which is not considered thermal conductors and heat escape.
3. Calculation and Verification

3.1. Calculation and Verification

According to the mass and energy release for postulated loss of coolant accident, the containment pressure and temperature was calculated and analyzed. In the paper, the pressure and temperature of containment with the free volume 1400m³ and 2000m³.

It was shown the calculation results of pressure changes in the containment of free volume 1400m³ with time in Figure 2, and it was shown the calculation results of steam and water temperature changes in the containment with time in Figure 3.

It could be seen from Figure 2 that the containment pressure reached the highest value at the end of spraying at 30 seconds, and the pressure between containment and the primary circuit was balanced.

The Fog.4 shown the calculation results of pressure changes in the containment of free volume 2000m³ with time, and the Figure 5 shown the calculation results of steam and water temperature changes in the containment.
3.2. Comparison of Calculation Results of Different Volumes

Free volume was the remaining space in the plant after deducing the volume occupied by the layout items. As a new reactor, the containment was been designed by conservative calculation and analysis. The change law of pressure and temperature in the containment with different free volume was calculated, as shown Figure 6 and Figure 7.

Figure 6. Change of Containment Pressure.
Figure 7. Change of Containment Temperature.

It was shown from Figure 6 and Figure 7 that with the increase of free volume, both of the containment temperature and pressure decreased, which could be used as the basis for designing the volume of the containment in consideration of the maximum pressure and temperature in case of DBAs. Increasing the free volume of containment was one of the measures to improve the safety of the nuclear power plant.

4. Conclusion
Free volume is an important input for the containment thermal analysis, and it provides a cushion space for the over-temperature and over-pressure after BDAs. It is assumed that the clod-leg double-end rupture accident will affect the temperature and pressure in the containment vessel with different free volume. This has certain guiding significance for the selection of the size of the containment vessel in actual engineering design.

References
[1] Gao Yingxian, Min Yuansheng, Chen Wei, et al. Research on Comparison of Containment Pressure and Temperature Response Based on PAREO and MELCOR. Science &Technology Vision, vol.2019, no.1, pp.217-220, 2019.
[2] Wang Guodong, Tang Weijian, Wang Zhe, et al. A Simplified Model to Simulate AP1000 Containment Pressure Response during Design Basis Accidents. Nuclear Power Engineering, vol.37, no.3, pp.163-168, 2016.
[3] Vijaykumar R, Khatib-Rahbar M. Applicability of the CONTAIN code heat and mass transfer models to asymmetrically heated vertical channels. Nucl. Technol. Vol.128, no. 3, pp.313-326. 1999.
[4] Tills J, Notafrancesco A, and Phillips J. Application of theMELCOR code to design basis PWR large dry containment analysis. Sandia National Laboratories, 2009, SAND2009-2858.
[5] Sha W T, Chien T H, Sun J G, et al. Analysis of large-scale tests for AP-600 passive containment cooling system. Nucl. Eng. Des.vol.232, no.2, pp.197-216. 2004.
[6] Yu J Y, Jia B S. PCCSAC: a three-dimensional code for AC600 passive containment cooling system analysis. Nucl. Sci. Eng.vol.142, no.2, pp. 230-236. 2002.