Containment Loads Analysis for CANDU6 Reactor using CONTAIN 2.0

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1. Introduction

The containment plays an important role to limit the release of radioactive materials to the environment during design basis accidents (DBAs). Therefore, the containment has to maintain its integrity under DBA conditions. Generally, a containment functional DBA evaluation includes calculations of the key containment loads, i.e., pressure and temperature effects associated with a postulated large rupture of the primary or secondary coolant system piping.

In this paper, the behavior of containment pressure and temperature was evaluated for loss of coolant accidents (LOCAs) of the Wolsong unit 1 in order to assess the applicability of CONTAIN 2.0 code [1] for the containment loads analysis of the CANDU6 reactor.

2. Methodology for Calculation

2.1 Analysis Tool

The CONTAIN 2.0 code was used for this analysis. The CONTAIN code was developed by Sandia National Laboratories (SNL) and is a specialized computer code used to perform thermal-hydraulic calculations inside containment following a variety of postulated high energy breaks. The CONTEMPT4/MOD4 code [2] was selected as a comparison tool.

2.2 Selection of the Accidents

In order to assess the applicability of CONTAIN 2.0 code for the CANDU6 reactor, 100% reactor outlet header (ROH) break accident and feeder pipe break accident of the Wolsong unit 1 were selected. The discharged mass and energy data for each accident were taken from the final safety analysis report for the Wolsong unit 1 [3].

2.3 Conditions of the Analyses

For our analysis, it was assumed that four out of six dousing headers were available and the first dousing spray started with delay of 7 seconds after the containment pressure reached the dousing spray actuation set point (2 psig).

The containment walls and the internal structures can act as energy sinks within containment. The walls and structures considered for the analysis included the perimeter wall, the internal wall, and internal structures such as columns, beams and steel lined walls and doors. Table 1 represents the features of the heat sinks. The mass and surface areas of these walls and structures were underestimated to minimize the heat sink.

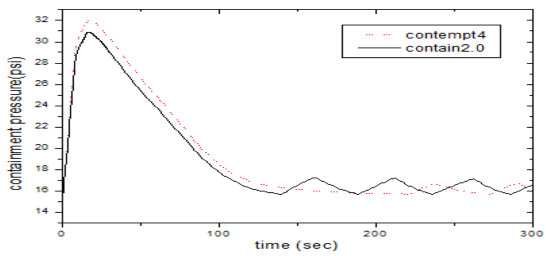
Table 1 Features of the heat sinks

	Surface Area (m^2)	Thickness (m)
Containment cylinder wall	4693	1.07
Containment dome	1353	0.6
Internal concrete walls	7648	0.5
Steel walls	223	0.01
Steel beams	8231	0.0253

3. Results of the Analysis

3.1 Containment Response of Reactor Outlet Header Break Accident

As shown in Fig. 1, the containment pressure and temperature are increasing rapidly at the early phase of the accident. The dousing spray starts at 8.2 seconds and its flow rate is built up to the design flow rate in 2.5 seconds. The peak pressure and temperature are appeared shortly after the onset of dousing spray and the pressure and temperature are decreasing thereafter.



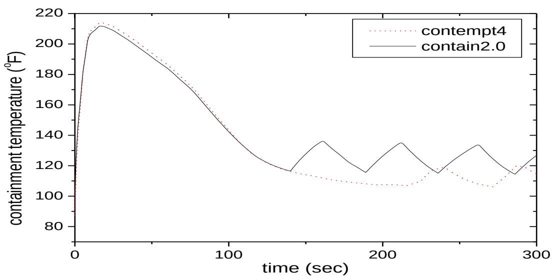


Fig. 1. Containment pressure and temperature of ROH break

The fluctuation of the pressure and temperature shown in the figure is because the dousing spray turns on when the containment pressure reaches 2 psig and off if it falls to 1 psig.

The peak pressure and temperature calculated by CONTAIN 2.0 are 30.9 psia and 211.8 ^{o}F, respectively. These values agree well with those of CONTEMPT4 calculation, 32.4 psia and 213.9 ^{o}F, respectively

3.2 Containment Response of Feeder Pipe Break Accident

The feeder pipe break is a small break loss of coolant accident. Therefore, the mass and energy discharge rates of the feeder pipe break are smaller than those of large break LOCAs. The dousing spray starts at 31 seconds and its flow rate is fully built up in 2.5 seconds in Fig. 2. Like the ROH break accident, the pressure and temperature reach their peak values right after the onset of dousing spray.

The peak pressure calculated by CONTAIN 2.0 and CONTEMPT4 are 17.2 psia and 17.4 psia, respectively. The difference of peak temperatures between two codes is 7.5 ^{o}F.

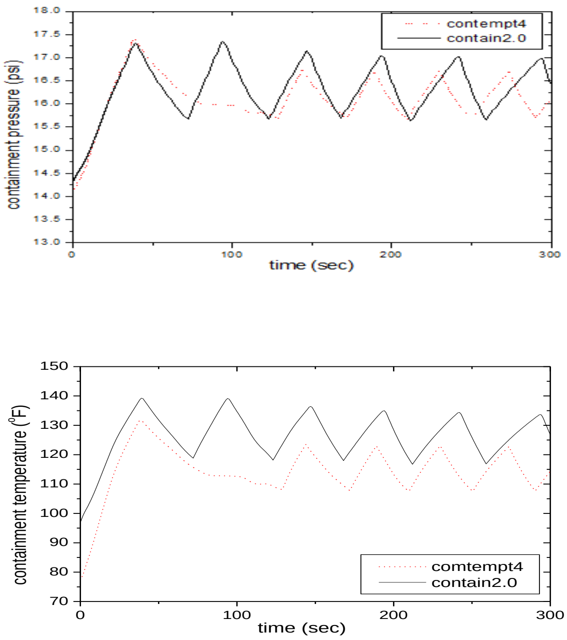


Fig. 2. Containment pressure and temperature of feeder break

3.3 Sensitivity Analysis of Dousing Delay Time

As you see in Fig. 1 and Fig. 2, the dousing start time is a crucial factor from the viewpoint of the containment peak pressure for CANDU6 reactor. In order to investigate the effect of dousing delay time on the containment peak pressure, five scenarios were assessed for the ROH accident. Table 2 presents the assumptions for each scenario and the sensitivity analysis result.

The result imply that it is important to shorten the dousing header filling time as well as the total dousing delay time to minimize the containment peak pressure.

Table 2 Effect of dousing delay time on containment peak pressure

	Scenario of dousing delay time	Peak Pr. (psia)
Case 1	Dousing flow is built up for 7.5s after dousing actuation signal	29.7
Case 2	Dousing headers are filled for 5s and dousing flow is built up for 2.5s after dousing actuation signal	30.3
Case 3	Dousing flow is built up for 9.5s after dousing actuation signal	30.0
Case 4	Dousing headers are filled for 7s and dousing flow is built up for 2.5s after dousing actuation signal	30.9
Case 5	Dousing flow is built up for 10.5s after dousing actuation signal	30.1

4. Conclusions

The containment pressure and temperature of the Wolsong unit 1 were evaluated using the CONTAIN 2.0 code and the results were compared with the CONTEMPT4 code. The peak pressure and temperature calculated by CONTAIN 2.0 agreed well with those of CONTEMPT4 calculation. The overall result of this analysis shows that the CONTAIN 2.0 code can apply to the containment loads analysis for the CANDU6 reactor.

REFERENCES

[1] Murata, K. K., et al., “Code Manual for CONTAIN 2.0: A Computer Code for Nuclear Reactor Containment Analysis,” NUREG/CR-6533, SAND-1735, Sandia National Laboratories, Albuquerque, NM, December 1997.

[2] Lin, C. C., et al., “CONTEMPT4/MOD4: A Multicompartment Containment System Analysis Program,” NUREG/CR-3716, BNL-NUREG-51754, Brookhaven National Laboratory, Upton, NY, March 1984.

[3] Final Safety Analysis Report for Wolsong unit 1, Korea Hydro & Nuclear Power Co., Ltd.