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Original Article

Simulation Of Containment Pressurization In A Large Break-Loss Of Coolant Accident Using Single-Cell and Multicell Models and CONTAIN Code

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ABSTRACT

Since the inception of nuclear power as a commercial energy source, safety has been recognized as a prime consideration in the design, construction, operation, maintenance, and decommissioning of nuclear power plants. The release of radioactivity to the environment requires the failure of multiple safety systems and the breach of three physical barriers: fuel cladding, the reactor cooling system, and containment. In this study, nuclear reactor containment pressurization has been modeled in a large break-loss of coolant accident (LB-LOCA) by programming single-cell and multicell models in MATLAB. First, containment has been considered as a control volume (single-cell model). In addition, spray operation has been added to this model. In the second step, the single-cell model has been developed into a multicell model to consider the effects of the nodalization and spatial location of cells in the containment pressurization in comparison with the single-cell model. In the third step, the accident has been simulated using the CONTAIN 2.0 code. Finally, Bushehr nuclear power plant (BNPP) containment has been considered as a case study. The results of BNPP containment pressurization due to LB-LOCA have been compared between models, final safety analysis report, and CONTAIN code's results.

1. Introduction

Nuclear safety has been one of the major issues to be studied since the inception of the nuclear industry. Nuclear reactor systems are sufficiently complex that dismissing the possibility of an accident followed by the release of radioactivity to the environment would be imprudent. Such a release would require the failure of multiple safety systems and barriers. Nuclear reactor containment is in fact the last of those barriers, and thus plays one of the main roles in nuclear safety.

The consequence of severe reactor accidents depends greatly on containment safety features and containment

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performance in retaining radioactive material. The early failure of the containment structures at the Chernobyl power plant contributed to the size of the environmental release of radioactive material in the accident. By contrast, the radiological consequences of the Three Mile Island Unit 2 accident were minor because the overall containment integrity was maintained and bypass was small [1].

During an accident in a water reactor nuclear power plant, the "blowdown" phase refers to the initial discharge, with a high mass flow rate of high-temperature pressurized coolant from the reactor cooling system into the containment. The intensity of the release is due to the high pressure difference between the cooling system and the containment atmosphere [2].

Given the importance of these issues, several studies have been carried out in recent years to evaluate the thermal–hydraulic behavior of the containment in an accident like large break-loss of coolant accident (LB-LOCA). In some cases, a valid code [CONTAIN, Generation of Thermal–Hydraulic Information for Containments (GOTHIC), etc.] has been used for this simulation, whereas in others a model has been developed for this purpose. Kljenak and Mavko [2] have simulated the thermal–hydraulic behavior of the containment in the Marviken Blowdown 16 experiment with the Accident Source Term Evaluation Code (ASTEC) and CONTAIN code. The GOTHIC and Reactor Excursion and Leak Analysis Program 5 (RELAP5) codes have been used by Papini et al [3] to analyze the International Reactor Innovative and Secure containment in a small break-LOCA (SB-LOCA). In addition, the GOTHIC code has been used to simulate SB-LOCA in the refurbished Wolsong-1 nuclear power plant by Kim and Park [4]. Finally, Ozdemir et al [5] have used this code for analysis of Fukushima Daiichi Unit 1 containment. The GOTHIC code is a general purpose thermal–hydraulic code used to model multicomponent and multiphase flow systems. This code is suitable for the safety analysis of nuclear power plant containment buildings [6]. Some other studies have been conducted using this code to simulate reactor containments [7–9]. In this study, in the first step, a single-cell model (total volume of containment has been considered as a control volume) has been developed to simulate containment pressurization. In the second step, the model has been developed into a multicell model and the effects of nodalization and the spatial location of cells have been considered on the results. These two models have been programmed in MATLAB. These programs can be used to study the different effects on containment pressurization in future work. In the third step, the CONTAIN 2.0 code has been used to simulate this accident. Finally, the results have been compared with Bushehr nuclear power plant (BNPP) final safety analysis report (FSAR) results.

2. Materials and methods

2.1. *Single-cell model*

The single-cell model has been developed according to Noori-Kalkhoran et al [6] by adding the spray model (see the "Spray Condensation" section) and wall condensation. The total volume of containment has been considered as a control volume. Mass and energy balance equations have been applied for this control volume. Some assumptions that have been applied in this model are as follows:

- The total volume of containment has been considered as a control volume.
- Containment includes three layers: steel containment, gap, and concrete containment.
- It is supposed that water from the break flashes to containment volume.
- The condensation layer has been considered in the containment inner side.

Fig. 1 shows the mass-, energy-, and heat-transfer processes in the single-cell model in the containment.

2.1.1. *Heat transfer*

Heat transfers from three layers according to the following equations prevent containment pressure from increasing beyond its design pressure.

2.1.1.1. Containment inner.

$$q_{\text{Heat Flux}} = h_{\text{conv}}(\Delta T) \quad \text{(Convection)} \tag{1}$$

where

$$h_{\text{conv}} = \left\{ 0.825 \left(\frac{9.8 \frac{1}{v_f} \left(\frac{1}{v_f} - \frac{1}{v_g} \right) k_f^3 i_{fg1}}{\mu_f (T_{\text{Cont}} - T_{\text{steel inner}}) l} \right)^{0.25} \right\} \tag{2}$$

and

$$i_{fg1} = i_{fg} + 0.68 C_{pf} (T_{\text{Cont}} - T_{\text{steel inner}}) \tag{3}$$

where h_{conv} is the convective heat-transfer coefficient in the containment in the presence of the condensation layer [10].

2.1.1.2. Steel and concrete layers. Heat transfers in the steel and concrete layers according to the conduction heat-transfer resistance equation are given as follows:

$$q_{\text{Heat Flux}} = \frac{\Delta T}{R} \tag{4}$$

where for the spherical shape, R should be as follows:

$$R = \frac{r_{\text{outer}} - r_{\text{inner}}}{4\pi r_{\text{inner}} r_{\text{outer}} k} \tag{5}$$

2.1.1.3. Gap and containment outer. Heat-transfer phenomena in the gap (between T_2 and T_3) and also between containment and environment (between T_4 and T_{air}) are natural convection processes. The convective heat-transfer coefficient of natural convection can be calculated according to the following equations [10]:

$$\text{Nu} = \frac{h_{\text{conv}} D}{k} = 2 + \frac{0.589 \text{Ra}^{1/4}}{\left[1 + (0.469/\text{Pr})^{9/16}\right]^{4/9}} \tag{6}$$

where

$$\text{Ra} = \text{Gr} \times \text{Pr} \tag{7}$$

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