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Simulation of early phase radioactivity of CPR1000 plant under LOCAs based on RELAP5-3D core engineering simulator

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ABSTRACT

In this study, the radioactivity of noble gases during loss of coolant accidents in containment is simulated by using CPR1000 nuclear power plant simulator in Ningde Fujian China. A simple fission product release model along with two real-time simulation methods are used for the modeling of the radioactivity transportation in the containment. In addition, an accurate method to simplify multi-nuclides into a single equivalent nuclide is presented. The characteristics of the lumped parameter method and the distributed parameter method for modeling containments are compared. Meanwhile, a shortcoming of the current containment modeling tool in the 3KeyMaster platform is discussed. The simulation results of noble gases gap release fractions are in agreement with the results of Sandia National Laboratories in SAND2008-6664 for high burnup cores.

1. Introduction

The radiation monitoring system (RMS) is an important system in nuclear power plants (NPPs), through which the radiation levels of the whole plant is monitored. It provides important information for diagnoses of the failure of barriers and early stage accident management. The simulation of RMS not only involves systematic modeling of radiation monitoring instruments, but also involves consideration of the core source term and some physical models. These models are related to 1) the release of fission products (FPs) from fuel rods into coolant, 2) the transportation of radionuclides in a system loop, 3) the release of FPs from system loop into containment, and 4) the transportation and removal of radionuclides in containment. In addition, the chemical forms of the nuclides are also considered. Therefore, the simulation of RMS is a result of the simulation of the radionuclides release and transportation processes. In summary, the essential tasks are to simulate the categories, timing, and fractions of nuclides released from the reactor core to the containment.

However, in many engineering simulators, the accuracy of RMS simulations under accident conditions is not satisfactory. There are two main reasons for this problem. First, the lack of measured radioactivity data of NPP under different kinds of accident conditions. Second, most of the available experiments and simulations tend to focus on considering the later stage source term of severe accidents without considering the early stage source term in detail. This leads to unexplored and inaccurate source term data that is required in the modeling of RMS in coolant activity phases and gap activity phases.

NUREG-1465 is an important report of source term for light-water nuclear power plants during accident. Using all the accident sequences identified in NUREG-1150 and performing corresponding STCP and MELCOR calculations, the NUREG-1465 presents the release fractions of the source term related to a typical timing sequence of different severe accident phases (USNRC, 1995). However, NUREG-1465 focus on fuel burnups less than 40 GW d/tU (USNRC, 1995). The previous physical models used in STCP and MELCOR are inadequate. Thus, NUREG-1465 is not suitable for current cores with higher burnups which are about 60 GW d/tU. In 2010, Sandia National Laboratories published SAND2008-6664 “Accident source terms for pressurized water reactors with high burn-up cores calculated using MELCOR1.8.5”, which can be used as an expanded reference to NUREG-1465. The results in SAND2008-6664 show that under accident conditions, the release fractions of noble gases, halogens and alkali metals in the gap activity phase are all less than 5% (Ashbaugh et al., 2010), regardless the different burnup cores. In fact, most of these release fractions are less than 3% (Ashbaugh et al., 2010). These release fractions in gap activity phases, plus early in-vessel release phases from PHEBUS-FP data

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are less than that of NUREG-1465 (Herranz and Clement, 2010).

In this study, 3KeyMaster platform from Western Service Company (WSC) is used as simulation environment. 3KeyMaster has been used in several nuclear power plant simulators such as Lungmen NPP's ABWR (Yang et al., 2012), Ningde NPP's CPR1000 and TerraPower's TWR. The platform can make graphical modeling of various NPP systems. Relap5-3D is integrated to further improve this platform. Unlike other versions of Relap5, Relap5-3D has integrated with the radionuclide transport model (INL, 2005). However, during the simulation of RMS, it has been found that there are some deficiencies in the platform's containment modeling tool which may affect the accuracy of simulation.

Presently, there are some specialized containment three-dimensional fluid dynamics analysis codes, such as GASFLOW (Nichols et al, 1998) and GOTHIC (Andreani and Paladino, 2010), which are used to analyze issues such as thermal hydraulics, hydrogen diffusion, and hydrogen combustion in containment under accident conditions (Kim and Hong, 2015; Papini et al., 2011). However, due to the requirement of real-time simulation, the three-dimensional fluid dynamics analysis codes mentioned above are not applicable. Considering the complexity of containment structures and the limitations of simulation tools, the accurate modeling of containments are very difficult in the simulation of RMS.

This work aims at 1) provide models and methods of RMS simulation which can be applied to engineering simulators, 2) study modeling methods of containment models to meet the real-time simulation requirement, and 3) analyze the deficiency of the containment modeling tool in 3KeyMaster. These efforts are made to improve the simulation accuracy of radiation monitoring system.

2. Mathematical physical models

2.1. Fission product release model

Fission product release model mainly focus on nuclides release processes and fraction release rates.

The release process of FPs in a fuel rod is divided into two steps. First, the FP is released from fuel pellets to a fuel-cladding gap, which is referred to as pellet release. Second, the FP is released from the fuel-cladding gap to coolant, which is referred to as gap release.

In general, most severe accident analysis codes, such as MELCOR, SCDAP and ASTEC, assume that all of the volatile FPs located in the fuel-cladding gap are released in a one-time release when the cracking of the cladding occurs at any axial elevation (USNRC, 2000, 1997). However, the gap release cannot release the entire initial inventory in the gap at the instant of cladding failure, though it is much faster than pellet release. In reality, the gap release takes a certain amount of time, not a one-time release.

After comprehensive consideration, a revised two-step method is proposed in this work, that is, 1) the release of FPs from the fuel rods, which is divided into gap release and pellet release; 2) at the instant of cladding failure, only a portion of the initial inventory of FPs in the gap can be released; 3) after that, the remaining FPs in the gap will be subsequently released along with the accumulated FPs in the pellet in accordance with specific release rate, such as CORSOR-M. The specific procedures of this method are as follows:

Assume the degree of cladding failure in the core region at the moment of t is:

$$\Delta\eta = \eta(t) - \eta(t-1) \tag{1}$$

where η is the degree of whole core cladding failure, and $0.0 \le \eta \le 1.0$.

The one-time release of nuclide mass from the fuel-cladding gap at the moment of t is:

$$m_\varphi = \varphi \cdot \theta \cdot M_i \cdot \Delta\eta \tag{2}$$

where M_i is the total accumulated mass of nuclide i in core in kg; θ is the initial mass fraction of nuclides in the fuel-cladding gap, for noble gases, the value is 0.03; φ is the defined mass fraction of nuclides of one-time release from gap inventory by using the revised two-step method, $0.0 \le \varphi \le 1.0$. The value of φ should base on accident type, nuclide type, the differential pressure of cladding break, the break size, and location. In this study the value 1.0 is set to model LOCA to meet the conservative assumption, and 0.1 is set to model a transient event of fuel rupture during normal operation.

During Δt, the period following the moment of t to the moment of next cladding failure, the available mass of nuclide released from fuel pellet is:

$$M_{av}(t) = (1 - \varphi \cdot \theta) \cdot M_i \cdot \Delta\eta + M_i \cdot \eta(t-1) - M_{re} \tag{3}$$

where M_{re} is the cumulative released mass of nuclide i by the end of pervious time step in kg. The value of M_{re} at each time step needs to be calculated and used for next time step.

The nuclide mass actually released from fuel rods during Δt is:

$$M_{release}(t) = m_\varphi + M_{av}(t) \cdot \left(1 - e^{-f \cdot \Delta t}\right) \tag{4}$$

where f is the fraction release rate of nuclide i from fuel pellet, its unit is s^{-1}.

Among many fraction release rate models, CORSOR-M model is popular for its comparatively simple and applicable to FPs with high, medium, low, and no volatility (USNRC, 2000). Therefore, from the perspective of simplification in an engineering simulator, CORSOR-M model is an appropriate choice. Its formula is as follows:

$$f(T) = K \cdot \exp(-Q/RT) \tag{5}$$

For noble gases, after setting model constants, their fraction release rates are expressed as

$$f = 2 \times 10^5 \exp\left(\frac{-63.8}{1.987 \times 10^{-3} T}\right) \tag{6}$$

where f is the fraction release rate, its unit is $\min^{-1}$; T is the average volume temperature of fuel pellet, its unit is K; R is the universal gas constant. The unit of constant K and Q is $\min^{-1}$ and kcal mol^{-1}, respectively.

2.2. The transport model used in system loop

The RELAP5-3D code has a built-in one-dimensional Euler transport model of radionuclides (INL, 2000), which assumes the radionuclide transported along with the flow of coolant in the pipeline. The corresponding radionuclide mass conservation equation is:

$$\frac{N_A}{M_W}\frac{\partial \rho}{\partial t} + \frac{1}{A}\frac{\partial}{\partial x}(CvA) = S \tag{7}$$

where N_A is the Avogadro constant; M_W is the atomic molar mass; ρ is the radionuclide mass density per unit volume; v is the fluid velocity; A is the cross sectional area in the direction of pipeline flow; S is the radionuclide sources (i.e. the number of radionuclide atoms emerged each second per unit volume).