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Simulation of the airborne radioactive substance distribution and monitoring of coolant leakage in a typical Nuclear Reactor Containment

Ya Zheng, Dongshuai Hu, Yiping Dai *

Institute of Turbomachinery, School of Energy and Power Engineering, Xi'an Jiaotong University, Xi'an 710049, China

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ABSTRACT

The management of coolant leakage in a Nuclear Reactor Containment is important to ensure the safe operation of a facility and preserve the structural integrity of the containment. This paper conducts the comprehensive analysis using the commercial Computational Fluid Dynamics (CFD) software CFX to assess the concentration distribution and motion of Xe-133 in a typical Nuclear Reactor Containment. Guidelines from the previous studies about the gas distribution in the containment are followed in the simulation. The containment model with detailed internal structure is established and divided into unstructured meshes. Analyses are carried out to quantify the Xe-133 distribution for three cases. These were (1) behavior analysis of Xe-133 for a given short period of time, (2) behavior analysis of Xe-133 for a given long period of time, (3) the effect of various leak rates and leak sources on Xe-133 distribution. The results indicate that the increasing rate of Xe-133 concentration is almost completely proportional to the leak rate of reactor coolant. The varying leak rates and leak sources have no significant impact on homogeneous distribution of leaking radionuclides in containment. The monitor of Xe-133 can ensure timely response to coolant leakage. This paper contributes to a reasonable design of the reactor coolant leakage detection system.

1. Introduction

The study of hydrogen diffusion and distribution is significant to plan the layout of dehydrogenation equipment and predict hydrogen concentration during accidental scenario. In the past three decades, several catastrophes, such as the Three Mile Island accident in 1972 and the Fukushima accident in 2011, have drawn much attention to this subject. As technology evolves, research and operating experience have indicated that very low levels of leakage could cause material degradation arising. For example, primary water stress-corrosion cracking and intergranular stress-corrosion cracking may be the result of boric acid corrosion. Such forms of degradation could potentially compromise the structural integrity of the containment and cause a loss-of-coolant accident (LOCA). Therefore, the monitoring of any leakage has also been put forward to locate its position.

According to the *Regulatory Guide 1.45* (Guide, 1973) published by U.S. NRC, the nuclear power plant (NPP) should use leakage detection systems with a response time (not including the transport delay time) of no greater than 1 h for a leakage rate of 1 gal/min (3.8 L/min). It is necessary to evaluate the leakage detection system, provide effective methods for monitoring leakage and identify its position. Reactor coolant normally contains sources of radiation that the monitoring systems can detect when these sources release to the containment. Monitoring changes of airborne substance radioactivity by an appropriate combination of leakage monitoring instruments can be an effective and indirect method for detecting a leak, as well as quantifying its flow rate. The effectiveness of airborne substance radioactivity monitors depends primarily on the activity of the reactor coolant and also, in part, on the containment volume and the background activity level. A survey of online pressurized water reactor (PWR) primary coolant leak detection technologies found that by monitoring the activity of noble gases, a power plant could detect the accurate leakage rate of coolant. Dissing et al. (1980) built a monitoring system based primarily on the detection of the N13 content in the containment atmosphere. Aoki (1991) evaluated the leak detection capabilities of several leakage monitoring systems in Japanese PWR plants. Goudarzi et al. (2015) calculated the fission products radioactivity in the primary coolant by using a set of ordinary differential equations which governs the fission products concentration in the primary coolant.

If the coolant leaks into the primary reactor containment, a variety of radionuclides can be found. Before designing an applicable

* Corresponding author at: Institute of Turbomachinery, Xi'an Jiaotong University, No. 28 Xianning West Road, Xi'an 710049, China. Tel./fax: +86 029 82668704.
E-mail address: ypdai@mail.xjtu.edu.cn (Y. Dai).

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Nomenclature

Abbreviation

CFD	computational fluid dynamics
DNS	direct numerical simulation
EVC	reactor pit ventilation system
EVR	containment continuous ventilation system
HVAC	heating, ventilation and air conditioning
LES	large eddy simulation
LOCA	loss of coolant accident
NPP	nuclear power plant
NRC	nuclear regulatory commission
PWR	pressurized water reactor
RANS	reynolds averaged Navier–Stokes
RCP	reactor coolant system
RRM	control rod drive mechanism cooling system
B_i	body force (N/m^3)
c_p	specific heat ($J/kg \cdot K$)
d, D	diameter, hydraulic diameter (m)
D_{nm}	equivalent binary diffusion coefficient for nth species
e	specific internal energy ($e = h - p/\rho$) (J/kg)
h	mean specific enthalpy (J/kg)
h_j^o	heat of formulation of the jth species at T_{ref}(kJ/kg)
k	turbulent kinetic energy (m^2/s^2)
L	characteristic length (m)
M	momentum source
m_n'''	source of nth species as the result of the chemical reactions ($kg/m^3 \cdot s$)
p	pressure (Pa)
q'''	heat generation per unit volume (W/m^3)
T	temperature (K)
T_{ref}	reference temperature (298.15 K)
x, y, z	coordinate directions

Symbols

Γ_e	mixture conductivity
η	viscosity coefficient
ρ	density (kg/m^3)
ω	species mass fraction

Subscripts

∞	free steam, bulk, ambient
g	gas phase
i	coordinate direction index
j, n	species index

Superscript

$'$	turbulent quantity (fluctuating component)
$'''$	volumetric source/sink terms

radioactivity monitoring system, an accurately quantitative relationship between the airborne radioactivity and the leak rate of coolant should be established. According to the previous studies reported in Bouhaddane and Farkas (2013), Rockwell III (1956) and Lin (1996), the total activation of any coolant may be broken up into three major subdivisions generally: intrinsic activity, corrosion products activity and fission product activity and the radionuclides in the primary circuit coolant mainly come from the following aspects: (1) the coolant, which refers to the nuclides with a short half-life generated by the radiated water molecules, such as O-16, N-16, N-13, O-17 and H-3, (2) the nuclear fission products. When the fuel cladding is damaged, nuclear fission products will be released into the coolant which could form almost 250 kinds of radioisotopes. The typical isotopes are Xe-135, Xe-133, Kr-85 and I-131 which all have a longer half-life, (3) the neutron-activated radionuclides, which are caused by some activated impurity in the coolant, like H-3, N-16, O-19 and Ar-41. And the neutron-activated radionuclides, which are produced by the reactor core structural materials activating, like Co-58, Mn-54 and Co-60.

Heitsch et al. (2010) performed a comparison of commercial CFD codes using GASFLOW, FLUENT and CFX for the Paks NPP. A defined severe accident scenario, which involves the release of hydrogen, was modeled. Simulations were performed on the full scale model of a typical Russian VVER 440-213 containment. All the codes showed comparable quantities of mitigated hydrogen, which was an integral quantity. Moreover, the investigation presented in the study in Baraldi et al. (2007) is a continuation of the numerical studies of validation and benchmarking. It must be emphasized that a first real-scale PWR model was already carried out by the numerical tools, after validation against small and large scale experiments (Baraldi et al., 2003). And it is the first calculation performed in order to demonstrate that the current codes and computer resources are capable of describing the real-scale phenomena in a nuclear containment.

In a previous work by Martín-Valdepeñas et al. (2007) where the CFD code FLUENT was first validated with relevant small scale benchmark problems. This step was also along the lines of the best practice guidelines for the use of CFD in nuclear safety applications (Mahaffy et al., 2007). The approach in this paper is somewhat similar to that of Prabhudharwadkar et al. (2011) where the FLUENT was used to assess the concentration distribution of hydrogen in a typical Indian Nuclear Reactor. The buoyancy modified Standard k–ε was found to be reasonably accurate and economical for prediction of hydrogen mixing problem involving large domain and computational mesh. Gyenes and Ammirabile (2011) analyzed the behavior of these aerosol particles what controls the fission-product transport into the containment and affects the global source term in a severe accident. A detailed thermal–hydraulic model was built to describe the aerosol distribution during the aerosol injection period.

The abundant research achievements have demonstrated the CFD method appear to be the most appropriate numeric tool to model the complex geometry of Nuclear Reactor Containment accurately. However most scholars focused on the hydrogen as the research object, and paid most attention to the severe accidents. And the study of the distribution and transport simulation of airborne radioactive substances, which is the potential hazard of power plant safety, had been ignored. In this paper, the real-scale geometry model of a typical Nuclear Reactor Containment including the detailed internal structure was established due to its huge impact on the reliability of the simulation. The following sections describe the containment and the leakage scenario, and the computer models developed for the particular codes. It is a demonstration of how CFD modeling is capable of providing significant indications for real-scale safety applications within the limits of uncertainty of the leakage scenario and helps to build an effective detection system for reactor coolant leakage.

2. Computational method

The airborne radioactive substance diffusion and distribution in Nuclear Reactor Containment is characterized by buoyancy-