Progress in Nuclear Energy 88 (2016) 75–87

Contents lists available at ScienceDirect

Progress in Nuclear Energy

journal homepage: www.elsevier.com/locate/pnucene

Modeling and simulation of radio-iodine released inside the containment as result of an accident

Khurram Mehboob [a, b, *], Mohammad S. Aljohani [a]

[a] Department of Nuclear Engineering, College of Engineering, King Abdul Aziz University (KAU), P. O. Box 80204, Jeddah 21589, Saudi Arabia
[b] Department of Physics, COMSATS Institute of Information Technology (CIIT), Chack Shahzad, Park Road, 44000, Islamabad, Pakistan

ARTICLE INFO

Article history:
Received 21 May 2015
Received in revised form 16 October 2015
Accepted 26 November 2015
Available online 29 December 2015

Keywords:
Iodine
LOCA
pH value
PWR
Spray flow rate
Spray droplet diameter

ABSTRACT

Significant amount of radioactive iodine could release from a commercial Nuclear Power Plant (NPP) in case of accident or due to rapture in primary loop piping. Quantification of iodine inside the containment is essential to study during normal or emergency situations of a NPP because it delivers the thyroid and lungs dose to workers in control room and in the surroundings. In this work, a semi kinetic model Iodine Activity Released in Containment air (IARIC) has been developed for parametric sensitivity study of iodine inside the containment. The model has been developed and implemented in FORTRAN 90 which contains deterministic and as well as a kinetic approach. The IARIC uses the isotope inventory evaluated by ORIGEN2 code (Croft, A. G., 1980) and isotopes data to determine the quantification of airborne iodine and its isotopes as a function of time. The sensitivity analysis of airborne iodine has been carried out as time dependent functions of recirculation rate, leakage rate, release rate with coolant, time dependent mixing rate in coolant. The effect of spray flow rate and spray droplet size has also been investigated. Kinetic simulation with continuous source of iodine as time dependent function of mixing rate *(w_x)* shows that w_x has a significant effect on airborne iodine activity. The iodine spray cleanup rate evaluated by IARIC are 3.22 h^{-1} and 6.68 h^{-1} which are in consistent with the rate found in literature.

1. Introduction and background

In order to determine the appropriate decision under the severe accident conditions of a NPP, the meticulous analysis of response of a NPP is essential to access the safety and risk margins. Owing to the strong influence of thermal hydraulic behavior, the fission products behavior is very complicated for release and transport under different accident sequences. The fission products behavior inside the containment is the fundamentals of source term. The source term results are the outputs of level 2 PSA (Denning et al., 1986), which are necessary for radiological assessments and consequences. Therefore, the behavior of fission products inside containment is essential to study under severe accidents.

During LOCA, radioiodine in gaseous and elemental form is one of the most important source term nuclide because of its high reactivity, high fission yield, environmental mobility, significant biological hazards and potential volatility in physical and chemical forms. It has complex chemistry, and it may transform into volatile species (Tigeras et al., 2011). Radioiodine is one of the most hazardous fission product in terms of reactor safety. Among iodine radioisotopes, the iodine 131 and 129 are the most significant isotope due to its large core inventory. ^{131}I has a half-life of 8 days and it delivers thyroid dose causing thyroid cancer in humans. Furthermore, iodine 129 has a half-life of 15.7 million year and it could became concentrated in thyroid glands and poses radiobiological hazards. Therefore, these risks are extensively studied under severe accident to design the needed technological barriers in NPP to minimize the environmental escape of iodine. The transportation of iodine in containment and its forms of release to the environment strongly depend upon its specification (Mehboob. K and Xinrong. C., 2012). During postulated LOCA conditions, iodine is available in reactor containment building as gaseous and fine elemental form (physical form). Since, iodine is relatively highly reactive as compared to other radioisotopes, it is also available in chemical forms. The iodine is available in reactors containment building in chemical forms as CH_3I, CsI, and HI (Soffer, L., et al., 1995).

* Corresponding author. Department of Physics, COMSATS Institute of Information Technology (CIIT), Chack Shahzad, Park Road, 44000, Islamabad, Pakistan. Tel.: +92 321 5144221, +966 550891690.
E-mail address: khurramhrbeu@gmail.com (K. Mehboob).

http://dx.doi.org/10.1016/j.pnucene.2015.11.013
0149-1970/

The radiation exposure from the NPP has been recognized as an important factor infusing both design of safety equipment and safety evaluation, including safety and risk assessment (Rosen. M., Jankowski. M., 1985). The determination of radiation hazards being evaluated for the development of nuclear technology. The Three-Mile Island (1979) and Chernobyl (1986) accidents diverted the research towards safety assessment and development of computational source term code packages. This research has resulted in development of several code packages (Mehboob. K., et al., 2012). The theoretical assessment results in several technical and safety reports (Mehboob. K, Xinrong. C., 2012). Besides theoretical assessment, numerous experiments have been performed to understand the unknown behavior of fission products (Lewis et al., 2008; Ducros et al., 2001; Luis and Herranz, 2010). Many experiments played an important role in understanding the behavior of fission products, iodine chemistry and transportation under accident situations (T. Hastet al, 2013; Luis and Herranz, 2010; N. Giraulta et al., 2012). Meanwhile, some R&D associations developed analytical codes (Mehboob, K et al., 2012). ASTEC is one of the popular code to study fission product behavior in severe accident situations (Kljenak et al., 2010). MELCOR with MACCS can be used to assess fission products release and to determine the radiological consequence (Haste et al., 2006). MAAP is a very popular computation code to calculate severe accident source term. Its quick calculation is its prime character.

Quantitative and qualitative analysis has been carried out for the source term determination for different NPPs for many years. However, Fukushima accident (2011) once more has diverted the research towards safety assessment and source term determination under severe accident scenarios. Some resent trends for source term analysis for Advanced Pressurized Water Reactor (APWR) has been carried out by (L.L. Tong, et al., 2015). Mehboob, K. et al. (2015) has quantified the in-containment source term for 1000 MWe PWR. Saeed E. Awan, at al. (2012) has studied the fission product behavior released inside the Pakistan Atomic Research Reactor-1 (PARR-1) containment by developing a kinetic model in FORTRAN. Lee and Ko (2008) has determined the source term for 3 loop PWR under severe accident situation. Mehboob, K. and Xinrong, C. (2012) have studied station blackout (SBO), LOCA, SBLOCA and Flow Blockage Accident (FBA) for 1000 MWe PWR and evaluated the source term for these situations. They have also measured the dependency of radioisotopes on containment safety systems, including containment spray and leakage rate (Mehboob, K., et al., 2013). Eslinger, P.W., et al. (2014) has quantified the radio xenon released into the atmosphere during Fukushima Dai-ichi accident. He also has computed the dispersion of radio-xenon in atmosphere released from Fukushima plants. Furthermore, Winiarek, V., et al. (2014) has quantified the ^{137}Cs source term caused from Fukushima Dai-ichi NPP accidents. He also has suggested an inverse dispersion model for atmospheric dispersion calculations, which depends upon atmospheric transport models. Yangmo, Z., et al. (2014) had used the (Mehboob. K, and Xinrong, C., 2012) model to perform simulation of hypothetical severe accidents at Saman nuclear power plant and has calculated the radiological dose. Recently, Younus and Yim (2015) has study the mitigation of out containment iodine source term by alkaline spray. Yunfei Zhao et al. (2015) has developed a system to determine the atmospheric source term for AP1000. L. Ammirabile et al., 2015 has used the ASTECV1.3 rev2 code to determine the source term for three loop 1000 MWe PWR under medium break LOCA (51.6 mm) and total electric power loss case (SBO). After Fukushima Dai-ichi accident Tianfeng Chaiet et al. (2015) has used Lagrangian dispersion model for the dispersion estimation of ^{137}Cs. Malet and Huang (2015) has studied the break-up of a light gas layer (helium) initially confined in the top of a closed volume. They have performed numerical simulation for helium concentration under spray. Moreover, they have performed several sensitivity studies to show the importance of the droplet size distribution, droplet velocity profiles, number of droplet classes, etc.

Even though, several studies for the source term have been carried out for the research reactors and for PWRs, the analytical prediction of iodine source term remains a challenge. The potential iodine source term in case of severe accident is highly influenced by the behavior of iodine within the containment, where iodine chemistry plays an important role. In this regards, NUREG 1465 (Soffer, L., et al., 1995) was the major mile stone that provides the most realistic estimation of source term released inside the containment building.

In this study, we have developed an analytical model to study the physical behavior of elemental iodine and quantification inside the containment as a result of an accident. For this reason, we have developed a semi-kinetic model to carry out the simulation study of airborne elemental and gaseous iodine inside the reactor building. Modeling and simulation of airborne iodine have been evaluated as a function of containment engineered mitigation processes with the continuous release of iodine from damaged fuel, the subsequent transport from fuel to coolant and from coolant to containment, with time-dependent mixing rate.

2. Simulation method

2.1. Characteristics of generation II PWR

A 2nd generation two loops Pressurized Water Reactor (PWR) coolant system models a Reactor Pressure Vessel (RPV), four coolant loops, two Steam Generators (SG), and a pressurizer connected in one of the loops. The primary system has two loops. Each loop has one hot leg, two Reactor Coolant Pumps (RCPs) and two cold legs and one pressurizer connected to one of the hot leg of one loop (Fig. 1). The key parameters of 1000 MWe generation II reactor are listed in Table 1. Light water is used as reactor's coolant and moderator as well. Secondary loop is modeled with main feed

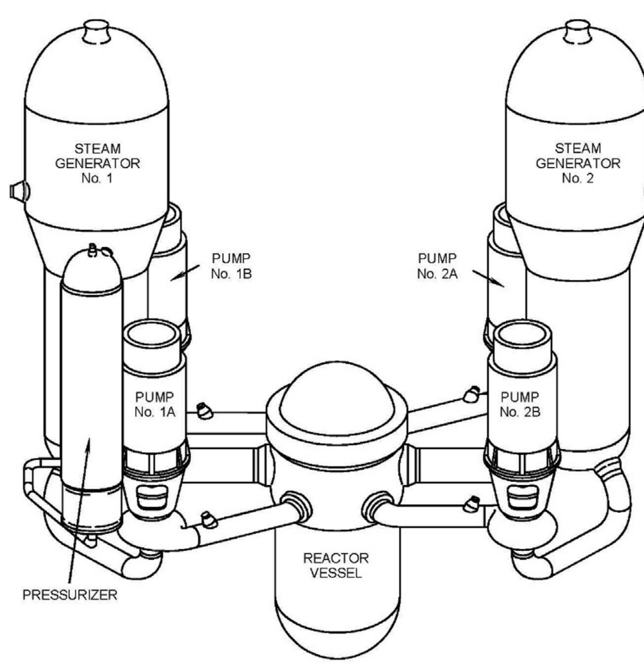

Fig. 1. A typical two loop PWR system.