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An analysis of radiological releases during a station black out accident for the APR1400

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ABSTRACT

An analysis on the behavior of fission products during a station blackout (SBO) accident for the APR1400 nuclear power plant is performed using MELCOR version 2.1. The analysis is focused on investigating the characteristics of radiological releases from the damaged fuel in the reactor core and molten corium in the reactor cavity, transport of the radioactive materials in the reactor coolant system and containment, and release of the radioactive materials to the environment. It is shown that the release characteristics of radionuclides from the fuel, distributions of radionuclides in the reactor coolant system and containment strongly depend on the species of radionuclides. In cases of volatile radionuclides such as iodine and cesium, more than 95% of initial inventory is released before the reactor vessel failure. The amount of radionuclides released to the environment after a failure of the containment was significantly reduced due to the deposition of radionuclides in the form of aerosols on the surfaces of heat structures and a pool of water in the containment. A sensitivity study on the presence of water in in-containment refueling water storage tank (IRWST) shows that the water in the IRWST significantly reduced the amount of radionuclides released to the environment.

1. Introduction

A Station Black Out (SBO) accident leading to a core damage and subsequent release of radioactive materials into the environment, which was evidenced in the Fukushima accident (Song and Kim, 2014; IAEA, 2015) highlighted the importance and the impact of the radiological releases during a severe accident on the public safety (Kim et al., 2017; Gupta, 2015). Fission products can be released from the damaged fuel in the reactor pressure vessel (RPV) and fuel debris located on the floor of the containment. In-vessel release is from the damaged fuel in the RPV during a fuel degradation and depends strongly on the scenarios of core melt progression (USNRC, 1990; Huang et al., 2010). The ex-vessel release commences when molten fuel debris is discharged from RPV to the reactor cavity after a failure of lower head. Radionuclides are released into the containment during the molten corium concrete interaction (MCCI) process (USNRC, 1990; Huang et al., 2015). The fission products could be transported to the containment before the failure of reactor vessel lower head through open sections of the reactor coolant system (RCS) such as pilot operated safety relief valves (POSRVs) or hot leg rupture. Subsequent leakage or failure of containment initiates environmental releases of fission products. In addition, fission products still can be released to the environment while the containment integrity is maintained during the containment bypass accident such as severe accident induced steam generator tube rupture accident (Rydl et al., 2016; USNRC, 2012).

The magnitude of the severe accident source terms depends on the plant design and the accident scenarios. NUREG-1150 (USNRC, 1990) estimated the source terms for five nuclear power plants in the USA. The source term analyses for several accident sequences for a typical pressurized water reactor (PWR) of Surry was performed in NUREG/CR-7110 (USNRC, 2012). Lee and Ko (2007) performed an analysis on the source terms for three-loop Westinghouse nuclear power plant. For a specific plant and accident sequence, there are still many factors which affect the source terms assessment, such as release mechanisms during core degradation within the vessel, during the interaction of the molten core with the concrete basement in the cavity, and the removal mechanisms of the fission product in the containment (USNRC, 2012). The ex-vessel release is influenced by type of concrete in the cavity, the core debris composition and temperature, and the presence of water in the cavity. The removal of radioactive materials in the containment can be performed by engineered safety systems such as containment spray, filter system and pool scrubbing. The deposition of aerosol on the structures surface and the water pool in the containment are considered as natural removal mechanisms in the containment (Kissane, 2008;

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NEA, 2000)

In this study, an analysis of the behavior of radioactive materials during station blackout (SBO) accident for the APR1400 is performed using MELCOR version 2.1 (Humphries et al., 2015). The Fukushima accident demonstrated the importance of SBO sequences among various scenarios of severe accidents (Kim et al., 2017). The objective of this study is to investigate the release and transport behavior of radioactive materials from the damaged reactor core to the containment and subsequent release to the environment. The amount of radioactive materials released to the environment would depend largely on the capacity of the containment in retaining radioactive material.

2. MELCOR modeling of the APR-1400

The APR1400 is a 1400 MWe two-loop pressurized water reactor (PWR) which has a large containment with a free volume of 95,000 m^3 (KEPCO, 2001). A schematic diagram and detailed description of a MELCOR modeling of the APR1400 is mentioned in detail in Vo and Song (2017). This paper addresses only important models relevant to the fission product behavior.

The core and lower plenum region is divided into a number of axial nodes and radial rings. Radial rings are numbered outward consecutively from the core centerline. Axial levels are numbered from the bottom to the top of the core. The core region is modeled as 6 radial rings including bypass region and 11 axial nodes from level 5 to level 15 where level 15 is the non-fuel region. Each ring in core channel region includes penetrations. The failure of lower head due to penetration failure occurs when temperature of penetrations reaches 1273.15 K. The failure of the lower head due to creep rupture occurs when the plastic strain in vessel lower head node reaches 18%.

Containment is subdivided into four control volumes, as shown in Fig. 1: (1) reactor cavity (CV001), (2) containment dome (CV009), (3) in-containment refueling water storage tank (CV012), and (4) main containment (CV002). The concrete thickness of the reactor cavity is 4.27 m at the bottom (KEPCO, 2001). The concrete type is limestone/common sand concrete and modeled as LIMESTONE/CS in the MELCOR (Humphries et al., 2015). The in-containment refueling water store tank (IRWST) is located inside the containment, which contains about 2500 m^3 of water (KEPCO, 2001). It plays a role as a primary heat sink for the discharge from the power operated safety and relief valves (POSRVs), as well as a water source for a safety injection pump, containment spray pumps, and cavity flooding system. In this analysis, all the engineered safety system are assumed to be unavailable due to SBO to provide the results of worst case scenario.

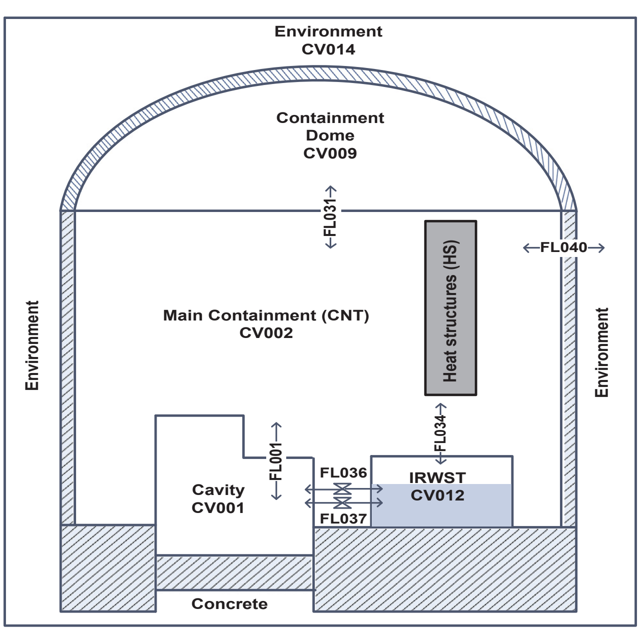

Fig. 1. Containment model in the MELCOR analysis.

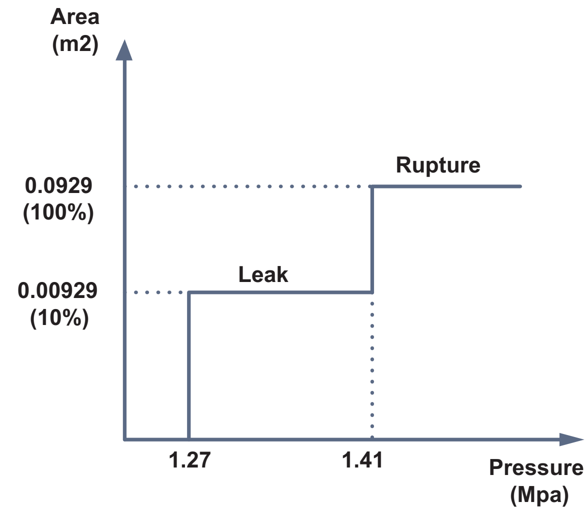

Fig. 2. Containment failure model in the MELCOR analysis.

The main containment (CV002) is the rest of the containment excluding the cavity, IRWST, and containment dome. The valve FL040 is used to model the connection between the containment and the environment. It opens when the containment pressure reaches failure conditions. A conceptual picture of the containment failure mode and opening area in the MELCOR analysis is illustrated in Fig. 2.

The heat structures consist of containment wall, dome, basement, and internal structures inside the containment. Specifications of the heat sinks in the containment are given in Table 1 (KEPCO, 2001). The heat structure for the dome belongs to control volume CV009. The IRWST upper slab is modeled as a heat structure of the IRWST (CV012). The basement is modeled as a heat structure for both the cavity

Table 1
Heat structures modeling in the containment.

No	Heat Sink	Materials	Thickness (m)	Surface area (m^2)
Containment Building				
1	Cylinder Wall	Carbon steel	0.00792	7572.6*
		Air	0.000151	
		Concrete	1.333	
2	Dome	Carbon steel	0.00664	3217.8*
		Air	0.000151	
		Concrete	1.067	
3	Basement	Concrete	13.934	392
Internal Concrete Structures				
4	Embedment	Carbone steel	0.0222	404
		Air	0.000151	
		Concrete	0.7898	
5	Unembedment	Concrete	0.7898	9695.5
6	Line fuel Pool	Stainless steel	0.00634	955.9
		Air	0.000151	
		Concrete	0.6858	
7	IRWST upper slab	Concrete	0.914	790
8	Polar crane & bridge	Carbon steel	0.0154	3679
9	Safety injection tank	Carbon steel	0.04737	506.6
10	Group A	Carbon steel	0.0162	6729.4
11	Group B	Carbon steel	0.00405	3294.1
12	Group C	Carbon steel	0.0146	2147.5
13	Group D	Carbon steel	0.006018	2332.3
14	Group E	Carbon steel	0.0026	1431.5
15	Group F	Carbon steel	0.00669	732.7
16	Group G	Stainless Steel	0.00635	1091

* The surface is calculated for one side.