Nuclear Engineering and Technology 53 (2021) 812–824

Contents lists available at ScienceDirect

Nuclear Engineering and Technology

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

Original Article

Analysis of fission product reduction strategy in SGTR accident using CFVS

Hoyoung Shin, Seungwoo Kim, Yerim Park, Youngho Jin, Dong Ha Kim, Moosung Jae*

Department of Nuclear Engineering, Hanyang University, 418 HIT, 222 Wangsimni-ro, Seongdong-gu, Seoul, 04763, Republic of Korea

ARTICLE INFO

Article history:
Received 7 February 2020
Received in revised form
12 July 2020
Accepted 3 August 2020
Available online 9 August 2020

Keywords:
Steam generator tube rupture (SGTR)
Fission product release
Severe accident management strategy (SAMG)
Power operated relief valve (PORV)
Containment filtered venting system (CFVS)

ABSTRACT

In order to reduce risks from the Steam Generator Tube Rupture (SGTR) accident and to meet safety targets, various measures have been analyzed to minimize the amount of fission product (FP) release. In this paper, we propose an introduction of a Containment Filtered Venting System (CFVS) connected to the steam generator secondary side, which can reduce the amount of FP release while minimizing adverse effects identified in the previous studies. In order to compare the effect of new equipment with the existing strategy, accident simulations using MELCOR were performed. As a result of simulations, it is confirmed that CFVS operation lowers FP release into the environment, and the release fractions are lower (minimum 0.6% of the initial inventory for Cs) than that of the strategy which intends to depressurize the primary system directly (minimum 15.2% for Cs). The sensitivity analyses identify that refill of the CFVS vessel is a dominant contributor reducing the amount of FP released. As the new strategy has the possibility of hydrogen combustion and detonation in CFVS, the installation of an igniter inside the CFVS vessel may be considered in reducing such hydrogen risk.

1. Introduction

A Steam Generator Tube Rupture (SGTR) which is an accident that one or more Steam Generator (SG) tubes break, is classified as a containment bypass accident because the coolant of primary system leaks to the secondary side through the break and fission products (FP) can be released into the environment directly. Therefore, SGTR is considered one of the most important accidents in terms of FP release. Also, the frequency of the initiating event of the accident in Pressurized Water Reactor (PWR) is 3.54×10^{-3}/rcry (Reactor Critical Year), and typically the Core Damage Frequency (CDF) is around 10^{-7} to 10^{-6}/rcry, which cannot be ignored [1]. Meanwhile, the Nuclear Safety Act in Korea revised in 2016 provides the following targets to be met in probabilistic safety assessment (PSA) for domestic Nuclear Power Plants (NPPs) [2]:

- Prompt and cancer fatalities among nearby residents due to the operation of and accidents at the nuclear power reactor facility shall satisfy 0.1% or lower of the sum of the total risk or the equivalent performance goal value. The total sum of accident frequencies that release the radionuclide Cs-137 over 100 TBq shall not exceed 1.0×10^{-6} /yr. In the Westinghouse Owners Group Severe Accident Management Guidance (WOG SAMG), severe accident mitigation strategies and their adverse effects are presented. In particular, Severe Accident Guidelines-05 (SAG-05) proposes the strategies regarding the release path of FPs. The strategies to minimize FP release when FPs are detected in the secondary side of SG, such as the SGTR accident, are as follows [3]: oolant System (RCS) to reduce FP release [4]. In the study, a SGTR accident in which all safety systems fail except passive systems in OPR1000 is selected as a reference accident scenario. It is assumed that the affected SG is successfully isolated, and the Main Steam Safety Valves (MSSV) are normally operated. As a result of analyzing the effectiveness of the strategy, the amount of FP released into the environment is reduced by 96.6–99.9% when Power Operated Relief Valve (PORV) opens after SAMG entry. Instead, the containment pressure increases by 51.5–60.0% with the valve opening.

There has also been a research to prevent FP release itself [5]. With an introduction of an In-Containment Relief Valve (IRCV) that connects with SG and free space in containment building, ICRV opens earlier than the safety valves of affected SG open. Therefore, FPs are dumped into the containment building rather than to the environment. Although there is no FP release by this

* Corresponding author.
E-mail address: jae@hanyang.ac.kr (M. Jae).

https://doi.org/10.1016/j.net.2020.08.003
1738-5733/

valve, there is an adverse effect of containment pressure rising to 1.2 MPa(a) which may threaten the containment integrity.

In this paper, we propose a Containment Filtered Venting System (CFVS) which is connected to the discharge line of MSSV and Pressure Relief Valve (PRV) on the SG main steam line. This lineup with CFVS can reduce the amount of FP release to the environment while minimizing the adverse effects identified in the previous studies, such as less efficiency of FP reduction, containment pressure build-up, and jeopardizing turbine building habitability. A direct depressurization strategy using PORV is also analyzed for the purpose of the comparison of FP release.

2. Research basis

2.1. Reference plant and accident scenario

Westinghouse two-loop (WH600) plant, which is one of the oldest operating NPP in Korea, is selected as a reference plant to evaluate the effect of the proposed strategy in terms of the FP release to the environment in the SGTR accident. WH600 is a PWR plant with thermal power of 1876 MW consisting of two SGs [6]. Table 1 shows the design characteristics of the reference plant.

SGTR-22 sequence is selected as a reference accident scenario among SGTR core damage sequences (refer to Fig. 1). It is a sequence that succeeds in controlling reactivity but fails in operating all active safety systems such as safety injection systems and secondary heat removal systems. Though Core Damage Frequency (CDF) of this sequence is less than 2% of the total CDF of SGTR scenarios, this sequence is expected to show the fastest accident progression among SGTR scenarios, because it loses the decay heat removal capability earlier than the other sequences. Therefore, this sequence is selected as a reference accident scenario despite its low contribution to CDF. In this paper, the reduction effect of FP release is analyzed assuming that a PRV of the affected SG is stuck-open after the initial opening.

2.2. MELCOR input model

MELCOR version 2.2 is used to simulate accident sequences. MELCOR, developed by Sandia National Laboratory (SNL) for the US Nuclear Regulatory Commission (NRC), is a comprehensive analysis code that can simulate the severe accident progress of Light Water Reactors (LWRs). It can model various severe accident phenomena such as thermo-hydraulic reactions in the RCS and in the containment building, core damage and relocation, Molten Corium-Concrete Interaction (MCCI), hydrogen behavior, and transport of fission products [7].

In order to reflect the geometry of the reference plant, the RCS and the containment building are divided into several Control Volumes (CV), and each CV is connected by a flow path (FL) to simulate material and energy transfer under severe accident conditions. Also, modeled are heat structures (HSs) for FP deposition and heat transfer in each CV, and core models for simulating heat and FP generation in the reactor core. The input has been developed to make it easier for analysts to prepare the desired accident scenarios using user-defined functions for trip or actuation signals of pumps and/or valves during the reactor operation. The data needed to develop the MELCOR input models referenced the Final Safety Analysis Report (FSAR) of the reference plant [6]. Fig. 2 shows the MELCOR nodalization input for RCS that reflects the design characteristics of WH600. In the nodalization, black letters represent control volumes, and red letters flow paths.

The accumulators (ACCs), the only source of cooling water in the reference scenario, are modeled as CV382 and CV482 (refer to Fig. 2). They supply coolant to cold legs when the pressure difference between the ACC and the cold leg exceeds 100 kPa(d). SG PRVs and MSSVs are modeled as valves in the SG secondary side. In SG train A (SG-A), where a SG tube breaks, the main steam pipe is divided into three control volumes (CV620, CV630, CV640) to consider FP deposition by gravity and turbulent flow in the pipe. CV620 and CV640 represent horizontal pipes, and CV630 a vertical pipe. Each node of main steam pipes reflects the design characteristics of the reference plant; pipe geometry, number of bends and curvatures [6].

The steady-state simulation shows a good agreement with the design operating conditions of WH600 presented in FSAR within 3% (refer to Table 2) [6].

3. FP release for reference scenario

MELCOR input assumptions for the simulation of the SGTR-22 sequence are shown in Table 3. It is assumed that a SG tube completely breaks at the bottom of SG-A (0.6 m above the SG inlet, refer to FL336 in Fig. 2), and its break area is 4.756×10^{-4} m^2 [8]. The SGTR occurs at 0 s, and the accident is analyzed for three days.

The reactor is shut down by the reactor trip signal. Since the SGTR-22 is a sequence for both the safety injection system and the secondary heat removal system to fail, it is simulated that all high-pressure safety injection pumps and auxiliary feed water supply to SGs fail. Containment Spray System (CSS) is also assumed to fail because SGTR is classified as a containment bypass accident. Passive safety systems, such as ACCs and Passive Autocatalytic Recombiners (PARs), are assumed to work normally. No design leakage of containment building is assumed.

SG PRV and MSSV open when the pressure of the SG secondary side exceeds 8114 kPa(a) and 8158 kPa(a), respectively, to release

Table 1
Plant design characteristics of WH600 [6].

Plant Design Characteristics	
System design Manufacturers	Westinghouse
Number of loops (Cold leg/Hot leg)	2/2
Thermal power	1876 MW
Reactor coolant system volume	181.9 m^3
Accumulator (ACC) coolant volume	70.8 m^3 (35.4 m^3/ACC $\times$ 2)
Containment free volume	40,193 m^3
Containment type	Large dry steel
Containment design pressure	410 kPa(a)[a]
Containment rupture pressure	920 kPa(a)
SG PRV opening set point	8114 kPa(a)
MSSV opening set point	8158 kPa(a)

[a] Absolute Pressure.

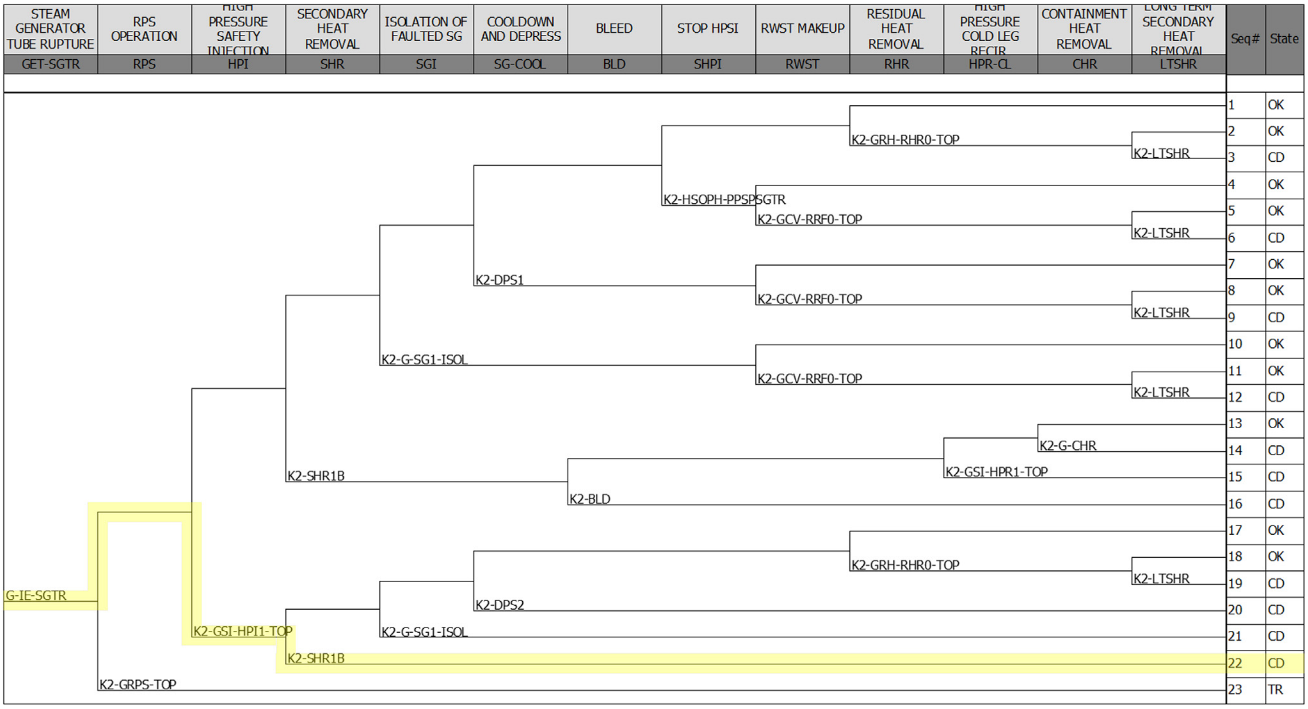

Fig. 1. SGTR event tree of reference plant.

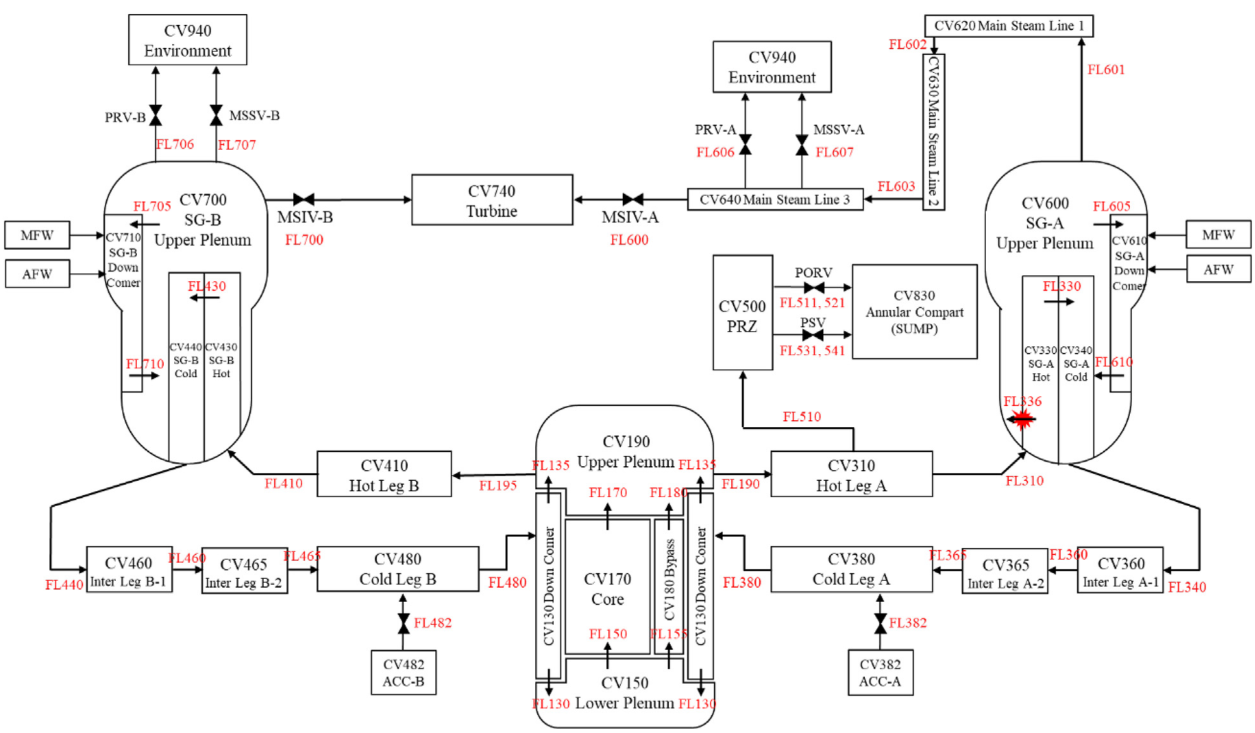

Fig. 2. MELCOR nodalization input of nuclear system for WH600.

steam from the main steam pipe to environment [6]. Since the reference scenario is a sequence that fails to isolate the affected SG (SG-A), it is assumed that one of SG-A PRVs is stuck-open after the initial opening. Therefore, it is simulated that FPs and coolant are leaking continuously from the primary side to the environment.

The timelines of significant event occurrences, the pressure of containment building after three days, and the release fractions of representative FPs (Cs, I, Te) are summarized in Table 4. In MELCOR, to evaluate the release and transport behavior of FPs, calculations are performed based on material classes grouping FP elements that have similar chemical properties [7]. In this paper, the release fraction of each radionuclide is calculated by dividing the total mass

Table 2
Comparison of key variables in steady-state [6].

	Unit	FSAR	MELCOR Calculation	Error [%]
Heat Power				
Core Heat Output	MW	1876	1876	0.0
Fission Power	MW	1755	1755	0.0
Decay Power	MW	120.6	120.6	0.0
Coolant Flow Rate				
Vessel Inlet	kg/sec	8840	8891	0.5
Core	kg/sec	8273	8426	1.8
Coolant Temperature				
Vessel Inlet	K	560.8	560.1	−0.1
Vessel Outlet	K	598.1	597.4	−0.1
Average Rise in Vessel	K	37.3	37.3	−0.1
Average Rise in Core	K	39.6	39.1	−1.2
Average in Core	K	581.7	579.7	−0.3
Average in Vessel	K	579.4	578.8	−0.1
Pressure Drop				
Core ΔP	kPa	182.7	187.5	2.6
Vessel ΔP	kPa	295.1	293.4	−0.5
SG ΔP	kPa	280.6	277.5	−1.1
Pressurizer				
Pressure	MPa	15.5	15.5	0.0
SG Secondary Side				
Steam Flow (per SG)	kg/sec	500.0	513.7	2.7
Feed Water Rate (per SG)	kg/sec	500.0	513.7	2.7
Feed Water Temperature at SG Inlet	K	494.4	494.4	0.0
Steam Pressure at SG Outlet	MPa	6.03	5.95	−1.2

of radionuclides released into the environment by the initial core inventory. At this time, the release fraction of Cs includes the amount of Cs nuclides in Cs compounds such as CsOH, CsI, and Cs2MoO4. Likewise, the fraction of iodine also includes the iodine of I_2 and CsI.

When the SGTR accident occurs, the primary coolant starts to leak to the secondary side due to the pressure difference between RCS and SG. It causes the RCS pressure down, as shown in Fig. 3, thereby shutting down the reactor and generating a safety injection signal. At the same time, Main Feed Water (MFW) pumps are stopped, and Main Steam Isolation Valves (MSIV) are closed. As Auxiliary Feed Water (AFW) systems fail, water left in SGs evaporates, the water level begins to drop (refer to Fig. 4, and a SG-A PRV is open at 922 s. As the affected SG PRV is stuck-open after the initial opening, the SG-A pressure continues to decrease. In SG-B, on the other hand, PRVs open and close when the secondary pressure reaches at their set points, keeping the SG pressure within the specified pressures. SG-B loses its water inventory after 66 h because less heat is removed through SG-B.

When the SG-A PRV is stuck-open, the affected SG pressure remains below the RCS pressure, until the Reactor Pressure Vessel (RPV) fails. As primary pressure drops below the ACC injection pressure, ACCs starts to inject cooling water at 1970 s. After ACC water exhaustion, the core water level gradually decreases, and the top of the core starts being uncovered. The Core Exit Temperature (CET) reaches the SAMG entry condition (922 K; 1200°F) at 11,706 s [9].

When the core temperature rises and cladding is damaged, FPs in the gap between fuel pellet and cladding begin to be released into the RCS, which is called "gap release." FPs leaked into the RCS are released into the environment through the broken SG tube and the stuck-opened SG-A PRV. The oxidation process on the cladding accelerates the fuel/cladding heat-up and relocation of the molten core to the lower plenum. Due to decay heat generated continuously, the lower head is heated up, and penetration failure occurs at 21,950 s (6.1 h).

Table 4
MELCOR simulation results for reference scenario.

Accident Progression	MELCOR Simulation Results
Reactor Trip, MFW Trip, MSIV Close [sec]	761 (0.21 h)
Reactor Coolant Pump Trip [sec]	781 (0.22 h)
SG-A PRV First (Stuck-) Open [sec]	922 (0.26 h)
ACC Start [sec]	1970 (0.55 h)
ACC End [sec]	8918 (2.48 h)
SAMG Entry [sec]	11,706 (3.25 h)
Cladding Failure [sec]	13,674 (3.80 h)
Reactor Pressure Vessel Failure [sec]	21,950 (6.10 h)
Containment Pressure at 72 h [kPa(a)]	253
Cumulative Release Fraction of Cs up to 72 h [−]	0.43
Cumulative Release Fraction of I up to 72 h [−]	0.61
Cumulative Release Fraction of Te up to 72 h [−]	0.65

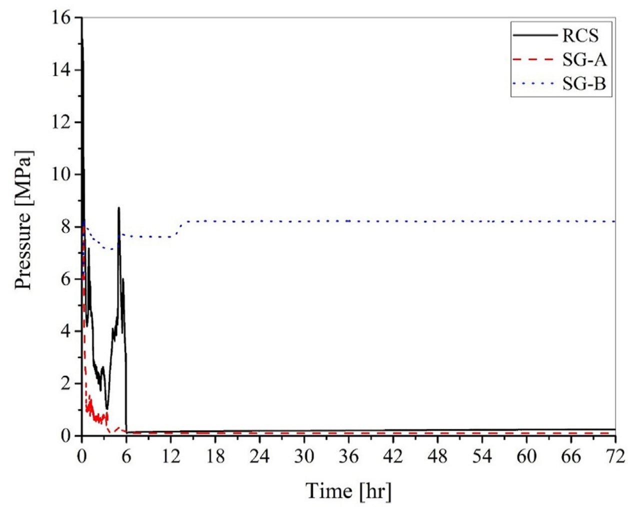

Fig. 3. Pressures of reactor coolant system and steam generator secondary sides for reference scenario.

Table 3
MELCOR inputs for reference scenario.

Descriptions	MELCOR Input Assumptions
Initiating Event	SGTR occurred at 0 s/Complete break of one of the tubes in SG-A (4.756×10^{-4} m^2)
Reactor Trip	Auto reactor trip by reactor trip actuation signal
High-Pressure Safety Injection	Fail to start HPSI Pumps
Secondary Heat Removal	Fail to deliver AFW to SG-A and SG-B
Containment Spray System (CSS)	Fail to start containment spray system
Accumulator (ACC)	Automatic operation when actuation conditions are satisfied
Passive Autocatalytic Recombiner (PAR)	Automatic operation when actuation conditions are satisfied
Isolation Failure of the Affected SG (SG-A)	SG-A PRV stuck-open after initial opening

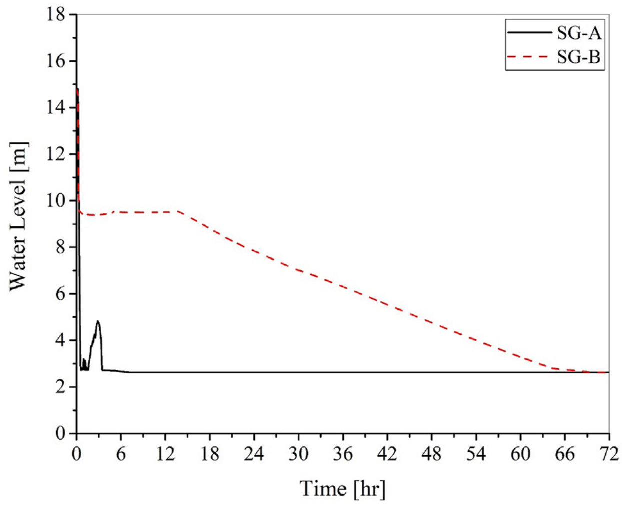

Fig. 4. Water level in steam generators for reference scenario.

Fig. 5 shows the mass distribution of Cs nuclide during the accident. Before the RPV penetration (6.1 h) fails, Cs released from the fuel by core degradation is transported into the RCS, SG secondary side, and environment. This is because the SGTR is a bypass sequence, and the FPs moved to the RCS are released to the SG secondary side and environment directly. In addition, the reference accident scenario has a large fraction (0.43 of the initial inventory for Cs nuclide) of direct release to the environment since the PRV of the affected SG is stuck-open.

When the vessel fails, molten corium is relocated to the reactor cavity. Following the pressure peak after vessel failure, containment pressure gradually rises due to steam and combustible gas generation at the reactor cavity by MCCI, reaching 253 kPa(a) three days after the accident (refer to Table 4). The amount of FPs released into the environment depends on the nuclide, and release fractions of Cs, I, and Te, which are representative nuclides that have significant effects on off-site consequence analysis, are estimated to be 0.43, 0.61, and 0.65, respectively (refer to Table 4 and Fig. 6).

4. Measures to reduce Fission Product Release using an existing equipment

Before evaluating the effectiveness of FP reduction using new equipment, the effectiveness of a measure using existing procedures is analyzed.

When SGTR accident occurs, cooling water as well as FP is leaking through the broken tube as long as there is a pressure difference between RCS and SG. The larger the pressure difference, the greater the amount of FP release. Therefore, the FP release can be minimized if RCS is depressurized by opening the pressurizer PORV. This strategy is described in SAG-02 of the reference plant [9].

The PORV generally performs a function to limit the RCS pressure below its set point (16.304 MPa(a)), but it can also be manually opened to depressurize the RCS directly [10]. The reference plant has two PORVs and the relieving capacity of each PORV at 16.304 MPa(a) is 95,254 kg/h [6]. Unlike a station blackout or other general transients, in a SGTR accident, a PORV may not open automatically because the RCS pressure decreases by leakage. Therefore, it is assumed that a PORV is manually opened after SAMG entry to reduce the pressure difference between RCS and SG.

In order to identify the FP reduction rate for the current SAMG strategy and identify an optimal method, a sensitivity analysis is performed with respect to two variables. First, the effect of the PORV opening time is analyzed. As the reference scenario, a PORV opens after SAMG entry, and it is 11,706 s (3.25 h) into the accident. For the sensitivity runs, a PORV open at intervals of 10 min–50 min. The second parameter is the number of open PORVs: one or two. The case that n PORVs open in m minutes after the SAMG entry is expressed as nP-m (number, minutes). For example, a case that one PORV opens 20 min after the SAMG entry is called 1P-20 case.

MELCOR simulation results for the sensitivity runs are summarized in Table 5 and Table 6.

4.1. Effect of PORV opening

Overall FP release trend is similar to the reference scenario except for the total amount of released FPs. The earlier the valve opens, the greater the reduction of FP is evaluated (refer to Table 5). This is because the earlier opening of PORVs reduces the flow rate from the RCS to the broken SG faster than the later opening cases.

It can be noted from Table 7 that the FP reduction rate for the nP-

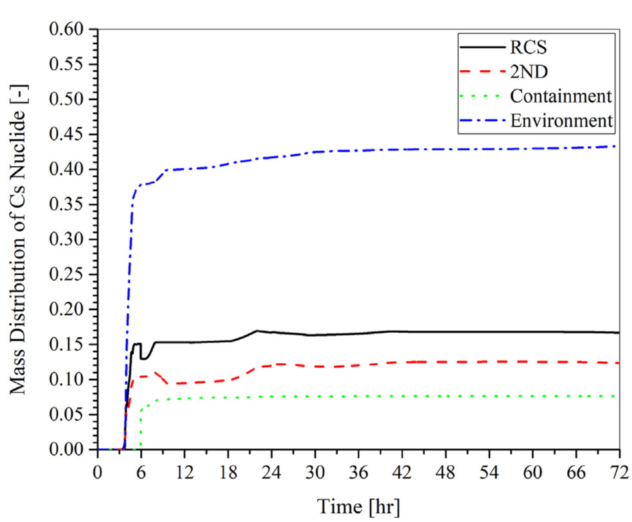

Fig. 5. Mass distribution of Cs nuclide for reference scenario.

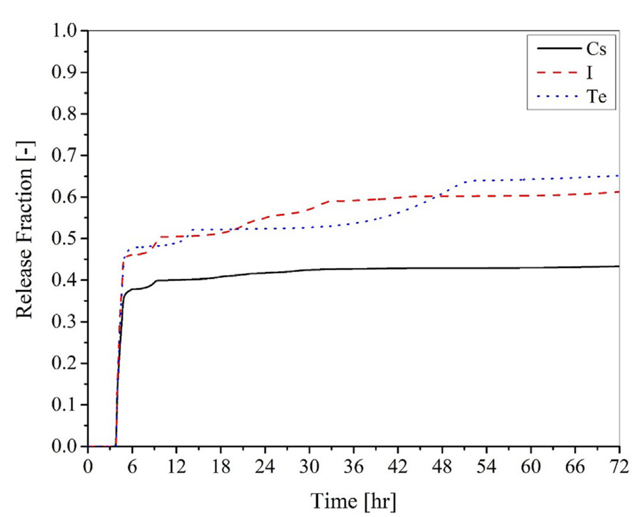

Fig. 6. Release fraction of representative radionuclides (Cs, I, te) for SGTR

Table 5
MELCOR simulation results for 1P- cases.

Accident Progress	MELCOR Simulation Results					
	Reference Scenario	1P-10	1P-20	1P-30	1P-40	1P-50
Accident progression before opening PORV is the same as the reference scenario (refer to Table 3)						
One PORV Open [sec]	-	12,306 (3.42 h)	12,906 (3.59 h)	13,506 (3.75 h)	14,106 (3.92 h)	14,706 (4.09 h)
Cladding Failure [sec]	13,674 (3.80 h)	13,724 (3.81 h)	13,720 (3.81 h)	13,682 (3.80 h)	13,674 (3.80 h)	13,674 (3.80 h)
RPV Failure [sec]	21,950 (6.10 h)	21,725 (6.03 h)	21,255 (5.90 h)	21,082 (5.86 h)	22,094 (6.14 h)	21,625 (6.01 h)
Containment Pressure at 72 h [kPa(a)]	253	297	294	295	293	288
Cumulative Release Fraction of Cs up to 72 h [−]	0.4333	0.2356	0.2347	0.2359	0.2942	0.3244
Cumulative Release Fraction of I up to 72 h [−]	0.6124	0.3293	0.3287	0.3303	0.4051	0.4466
Cumulative Release Fraction of Te up to 72 h [−]	0.6515	0.2206	0.2278	0.2315	0.3317	0.4297

Table 6
MELCOR simulation results for 2P- cases.

Accident Progress	MELCOR Simulation Results					
	Reference Scenario	2P-10	2P-20	2P-30	2P-40	2P-50
Accident progression before opening PORV is the same as the reference scenario (refer to Table 3)						
Two PORVs Open [sec]	-	12,306 (3.42 h)	12,906 (3.59 h)	13,506 (3.75 h)	14,106 (3.92 h)	14,706 (4.09 h)
Cladding Failure [sec]	13,674 (3.80 h)	13,785 (3.83 h)	13,854 (3.85 h)	13,784 (3.83 h)	13,674 (3.80 h)	13,674 (3.80 h)
RPV Failure [sec]	21,950 (6.10 h)	21,012 (5.84 h)	21,444 (5.96 h)	21,505 (5.97 h)	21,574 (5.99 h)	22,018 (6.12 h)
Containment Pressure at 72 h [kPa(a)]	253	302	303	298	299	291
Cumulative Release Fraction of Cs up to 72 h [−]	0.4333	0.1613	0.1523	0.1526	0.2410	0.2841
Cumulative Release Fraction of I up to 72 h [−]	0.6124	0.2110	0.2274	0.2112	0.3270	0.3918
Cumulative Release Fraction of Te up to 72 h [−]	0.6515	0.1423	0.1366	0.1368	0.2277	0.2979

Table 7
Reduction rate of FP release compared to reference scenario for representative radionuclides (PORV sensitivity cases).

Cases	Cumulative Release Fraction up to 72 h [−]			Reduction Rate [%]		
	Cs	I	Te	Cs	I	Te
1P-10	0.2356	0.3293	0.2206	45.6	46.2	66.1
1P-20	0.2347	0.3287	0.2278	45.8	46.3	65.0
1P-30	0.2359	0.3303	0.2315	45.5	46.0	64.4
1P-40	0.2942	0.4051	0.3317	32.1	33.8	49.0
1P-50	0.3244	0.4466	0.4297	25.1	27.0	34.0
2P-10	0.1613	0.2110	0.1423	62.7	65.5	78.1
2P-20	0.1523	0.2274	0.1366	64.8	62.8	79.0
2P-30	0.1526	0.2112	0.1368	64.7	65.5	79.0
2P-40	0.2410	0.3270	0.2277	44.3	46.6	65.0
2P-50	0.2841	0.3918	0.2979	34.4	37.6	54.2

10 to nP-30 is similar to each other but higher than the case of nP-40 and nP-50. This trend is related to the cladding damage timing. When the PORVs open after the cladding failure (~13,674 s), most of the FPs released from the fuel (cladding) will escape to the broken SGs, causing less FP trapping in the containment.

This suggests that implementing the reference strategy within 30 min after SAMG entry is effective in terms of FP reduction.

4.2. The number of opening PORV

As a result of sensitivity analyses on the number of opened PORV, the FP reduction rate is higher when both PORVs open (refer to Table 7). This is because, as two PORVs open, the area of the flow path from the primary side to containment building is enlarged, so that the RCS pressure drops more quickly. The release fractions when both PORVs open are reduced to 64–88% compared to 1P- cases for Cs nuclide.

The cases that both PORVs open within 30 min after SAMG entry is the most effective ways to reduce FP release. In the case of 2P-20, which shows the highest FP reduction rate, the reduction rate of Cs is estimated to be 64.8% compared to the reference scenario.

4.3. Adverse effects

The opening of PORVs will cause containment pressure build-up. When one PORV opens, the pressure increases by 13.7–17.3% compared to the reference scenario, and 14.9–19.5% when both PORVs open (refer to Table 8). Containment pressure behavior for 72 h in 2P- cases are shown in Fig. 7.

However, even the 2P-20 case shows the highest containment pressure of 303 kPa(a) 3 days after the accident. It is far below the containment rupture pressure (920 kPa(a)). Besides, considering that the minimum pressure at which the probability of containment damage occurs from the containment fragility curve of reference plant is about 700 kPa(a) [10], the adverse effect due to pressure rise in the containment is negligible.

5. Measures to reduce Fission Product Release using new equipment

Containment Filtered Venting System (CFVS) is originally designed to prevent containment over-pressure by venting air in containment to the environment [11]. Even though the venting may be necessary as the last resort of preventing containment over-pressurization, it accompanies the undesirable off-site release of FP. So filtration function is added to minimize the amount of FPs emitted. In general, filtration function is implemented by spargers and mechanical filters. Fig. 8 shows a typical schematic diagram of the reference CFVS. A sparger is a device that sprays gas into the liquid, and the containment air is jetted into the water pool in the CFVS vessel by the sparger. In this process, water-soluble FPs are primarily removed. But some gaseous FPs, such as elemental and organic iodine, and the unscrubbed aerosol-type FPs from the sparger pool are trapped by the filter [12].

In this paper, we focus on the filtration function of CFVS and propose its application in the SGTR accident, though CFVS is currently designed to prevent the containment from over-pressurization. If three-way valves (hereinafter referred to as CFVS valves) are installed downstream of SG PRV and MSSV, steam containing FPs in the main steam pipe could be delivered into the

Table 8
Pressure rise of containment building compared to reference scenario (PORV sensitivity cases).

Cases	Containment Pressure at 72 h [kPa(a)]	Pressure Rise Rate of Containment Building [%]
1P-10	297	17.3
1P-20	294	15.9
1P-30	295	16.4
1P-40	293	15.5
1P-50	288	13.7
2P-10	302	19.2
2P-20	303	19.5
2P-30	298	17.8
2P-40	299	18.0
2P-50	291	14.9

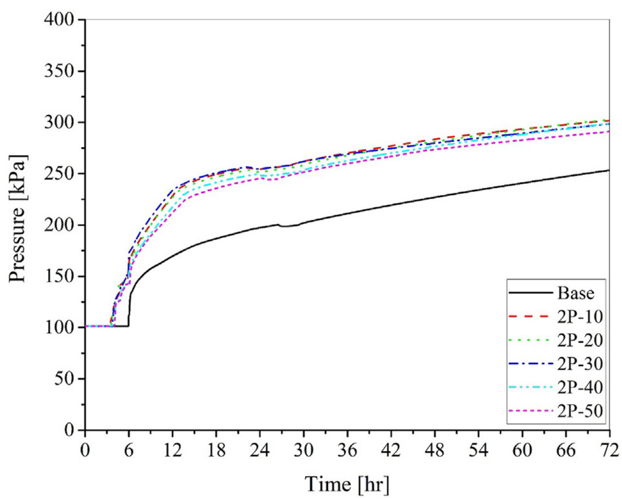

Fig. 7. Containment pressure for reference scenario and 2P- cases.

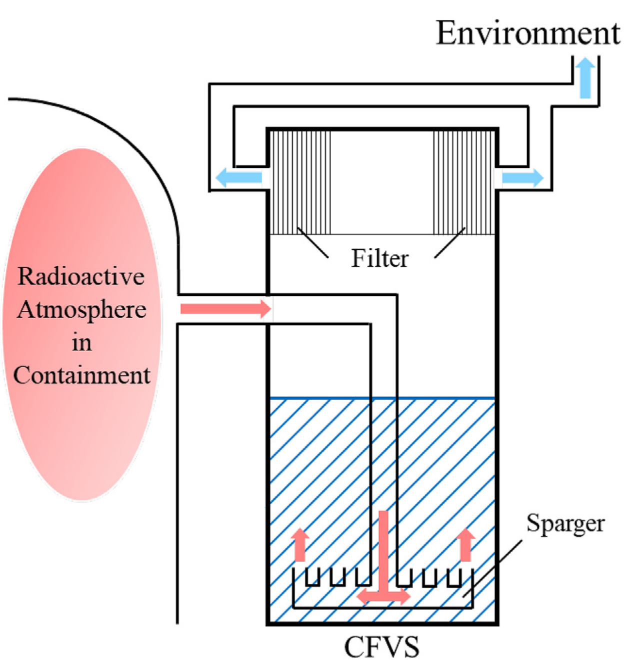

Fig. 8. Schematic diagram of reference containment filtered venting system.

CFVS vessel and filtered out to reduce FP emissions to the environment in the SGTR accident (refer to Fig. 9).

CFVS should be designed implementing the operating conditions such as CFVS inlet pressure, flow rate, temperature and gas composition. Each operating condition depends on the design characteristics of the NPP. However, CFVSs are not installed in Korean PWRs at present. Therefore, AREVA's CFVS reflecting the design characteristics of the reference NPP, is assumed to be installed as a reference equipment. It has a cylindrical CFVS vessel of 6.5 m in height and 3 m in diameter. The height of the sparger is 1 m from the bottom of the vessel, and the water level of the pool is 3 m [13,14]. The diameter of pipe connecting SG and CFVS vessel is assumed to be equal to that of PRV and MSSV line.

The SPARC model of MELCOR is used to estimate the effect of the sparger. This model is used to perform FP scrubbing calculations in the CFVS pool [7]. The sparger of the reference CFVS consists of 24 holes in 6 arms [12], and the diameter of each hole is assumed to be 10 mm.

Apart from the pool scrubbing function, the AREVA's CFVS offers two other filtering functions; aerosol and vapor iodine filtering. Decontamination Factor (DF) of the aerosol filter is found to be 10 from other papers [15]. Though the vapor iodine filter is classified into an organic iodine filter and elemental iodine filter, the conservative (lower) value of DF 5 is used as a representative DF for vapor iodine filter [14]. The data of CFVS required as input to MELCOR are summarized in Table 9.

In this study, it is assumed that a flow path to CFVS is connected by opening the CFVS valve after SAMG entry. That is, before SAMG entry, steam exits to the environment directly, but after SAMG entry, it will be released via CFVS. In the reference accident scenario, since the PRV of the affected SG is stuck-open, steam continuously flows into the CFVS through PRV after the SAMG entry condition is satisfied.

Several sensitivity analyses are performed to identify differences in the FP reduction rate implementing the new SAMG strategies. Table 10 summarizes the list of sensitivity analysis cases:

- C-Base: CFVS Base case (A case that does not refill CFVS vessel)
- FR-m (Flow Rate): Cases with a refill rate to CFVS vessel (m; mass flow rate [kg/sec])
- WL-n (Water Level): Cases with minimum water level to start refill (n; minimum water level in CFVS vessel [m])
- T-k (Time): Cases with CFVS valve opening time after SAMG entry. (k; time [minutes])

5.1. CFVS Base Case

The C-Base is a case that the CFVS valve is operated 10 min into the SAMG entry, and the CFVS vessel is not refilled even after the coolant in the vessel evaporates completely. MELCOR results for the C-Base case are shown in Table 11. As this paper is not intended to design CFVS for the SGTR, it is assumed that the CFVS was designed as necessary for its function. The calculated boundary conditions of the main steam line and the CFVS inlet at the time of the CFVS actuation are as follows. The pressure of the main steam line is 345 kPa(a), and the CFVS pressure keeps the atmospheric pressure. The vapor temperature in the main steam line and the CFVS are 563.9 K and 290.4 K, each. The flow rate at the CFVS inlet connected to the SG-A RPV is 4.4 kg/s. The containment pressure increases by 2.3% compared to the reference scenario. FP reduction rates of Cs, I, and Te nuclides for the reference scenario are 54.1%, 56.8%, and 66.7%, respectively (refer to Table 16). The reason for the decrease of FP release is, therefore, obvious. It is because FPs, which are previously released directly to the environment, are now scrubbed,

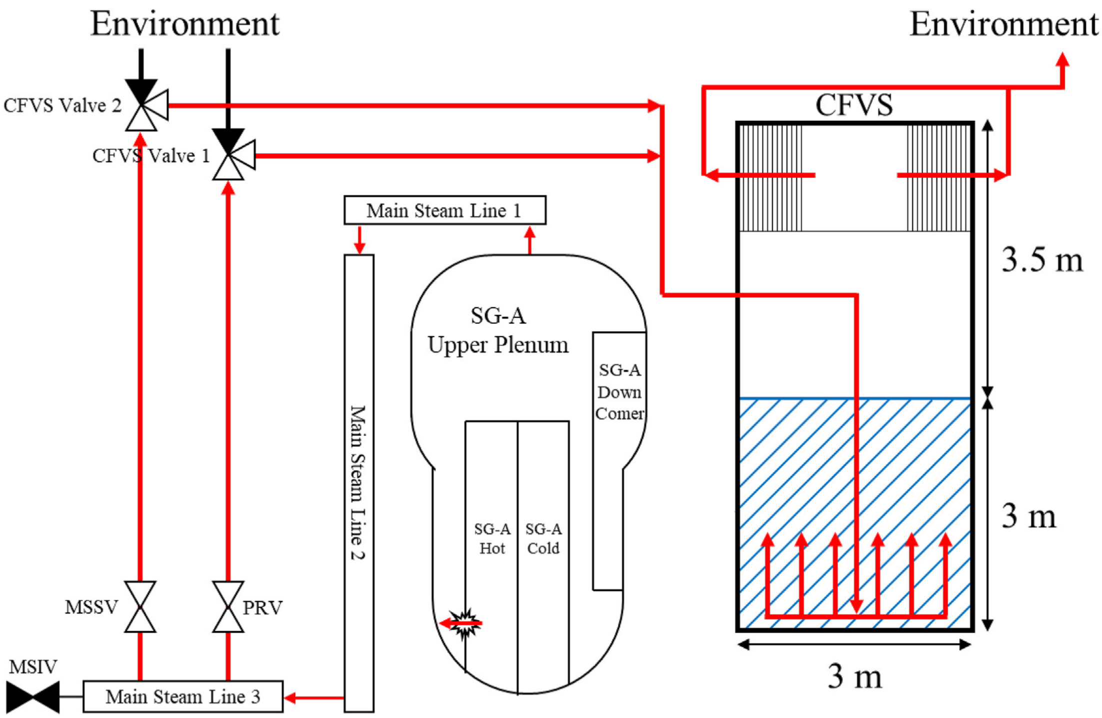

Fig. 9. Schematic diagram of new path of fission product release in SGTR accident.

Table 9
CFVS design data for MELCOR modelling [13,14].

CFVS Data		
CFVS Vessel	CFVS Vessel Height [m]	6.5
	CFVS Vessel Bottom Diameter [m]	3
	Pool Height [m]	3
Pipe	PRV to CFVS Pipe Diameter [m]	0.112
	MSSV to CFVS Pipe Diameter [m]	0.107
Sparger	Diameter of hole [m]	0.01[a]
	Number of holes	24
Filter	Decontamination Factor for Aerosols	10[a]
	Decontamination Factor for Vapor Iodine	5

[a] Assumptions.

Table 10
CFVS sensitivity cases.

Sensitivity Cases	Refill Rate [kg/sec]	Water Level to Start Refill [m]	Time to Operate CFVS Valve After SAMG Entry [min]
C-Base	-	-	10
FR-0.1	0.1	1.5	10
FR-0.5	0.5		
FR-1	1		
FR-10	10		
WL-1.5	10	1.5	10
WL-2.0		2.0	
WL-2.5		2.5	
WL-3.0		3.0	
T-10	10	1.5	10
T-20			20
T-30			30
T-40			40
T-50			50

filtered, and discharged through the CFVS.

Fig. 10 shows the release fraction of Cs nuclide for the C-Base and FR-10. The release fraction of the C-Base case is negligible until 13.3 h, but it suddenly increases after water in CFVS is depleted. Immediately after CFVS operation, the vapor condenses in the pool and the CFVS water level rises. However, due to the high-temperature steam released from the SG secondary side and the decay heat from FPs in the water pool of the CFVS vessel, the CFVS pool temperature reaches saturation temperature in 4.7 h (16,900 s) and starts boiling (refer to Fig. 11). When the water in the vessel is exhausted, no more reduction in FP emissions by scrubbing is expected.

Another reason for the increase in FP release after water depletion is the revaporization of the FPs, which have been deposited on the surface in the CFVS vessel. As the surface temperature increases, FPs deposited on the surface revaporize [16]. Fig. 12 shows the mass of FPs in the gas phase, in the liquid phase, and deposited on the CFVS vessel inner surfaces. Before water depletion occurs at 13.3 h, FP is trapped in the pool and mostly exists in liquid phase. However, when the CFVS water pool is depleted, FPs dissolved in the liquid are left on the CFVS inner heat structures. Afterwards, a part of the FPs deposited on the heat structures (up to 73%) are revaporized and released into the environment through the filter. So even though CFVS filter works normally, the amount of FP released after water depletion increases.

Therefore, in order to enhance the efficiency of FP reduction, refill of the CFVS vessel will be necessary to maintain pool scrubbing function. Sensitivity analyses are conducted to identify the effective refill strategies (see sections 5.3 ~ 5.5).

5.2. Effect of CFVS refill

C-Base is compared with FR-10 that starts refill at a flow rate of 10 kg/s when the water level of the CFVS vessel pool reaches 1.5 m, providing a sufficient flow rate to maintain the water level of the CFVS. Therefore, FR-10 is compared with the C-Base case to analyze the effect of water depletion in the CFVS vessel on the FP reduction. Table 12 shows the simulation results for both cases. In the case of FR-10, the release fraction is reduced to about 1/30 of the C-Base

Table 11
MELCOR simulation results for reference scenario and C-Base case.

Accident Progress	MELCOR Simulation Results	
	Reference Scenario	C-Base
Accident progression before operating CFVS valve is the same as the reference scenario (refer to Table 3)		
CFVS Valve Operation [sec]	-	12,306 (3.42 h)
Cladding Failure [sec]	13,674 (3.80 h)	14,123 (3.92 h)
RPV Failure [sec]	21,950 (6.10 h)	21,530 (5.98 h)
Containment Pressure at 72 h [kPa(a)]	253	259
Cumulative Release Fraction of Cs up to 72 h [–]	0.4333	0.1985
Cumulative Release Fraction of I up to 72 h [–]	0.6124	0.2641
Cumulative Release Fraction of Te up to 72 h [–]	0.6515	0.2168

Table 16
Reduction rate of FP release compared to reference scenario for representative radionuclides (CFVS sensitivity cases).

Cases	Cumulative Release Fraction up to 72 h [–]			Reduction Rate [%]		
	Cs	I	Te	Cs	I	Te
C-Base	0.1985	0.2641	0.2168	54.1	56.8	66.7
FR-0.1	0.0269	0.0811	0.0638	93.7	86.7	90.2
FR-0.5	0.0067	0.0361	0.0169	98.4	94.1	97.4
FR-1	0.0066	0.0358	0.0163	98.4	94.1	97.4
FR-10	0.0063	0.0354	0.0159	98.5	94.2	97.5
WL-1.5	0.0063	0.0354	0.0159	98.5	94.2	97.5
WL-2.0	0.0064	0.0355	0.0158	98.5	94.2	97.5
WL-2.5	0.0064	0.0353	0.0157	98.5	94.2	97.5
WL-3.0	0.0064	0.0353	0.0156	98.5	94.2	97.6
T-10	0.0063	0.0354	0.0159	98.5	94.2	97.5
T-20	0.0140	0.0377	0.0152	96.7	93.8	97.6
T-30	0.0260	0.0483	0.0232	94.0	92.1	96.4
T-40	0.1261	0.1983	0.1509	70.9	67.6	76.8
T-50	0.1909	0.3163	0.2588	55.9	48.3	60.2

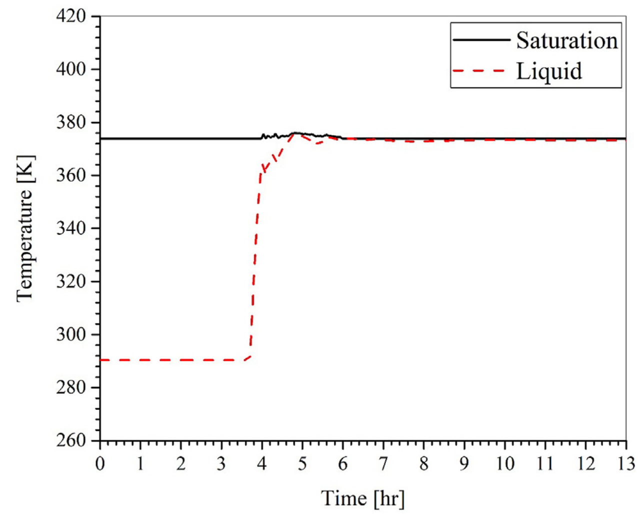

Fig. 11. CFVS pool temperature for C-Base case.

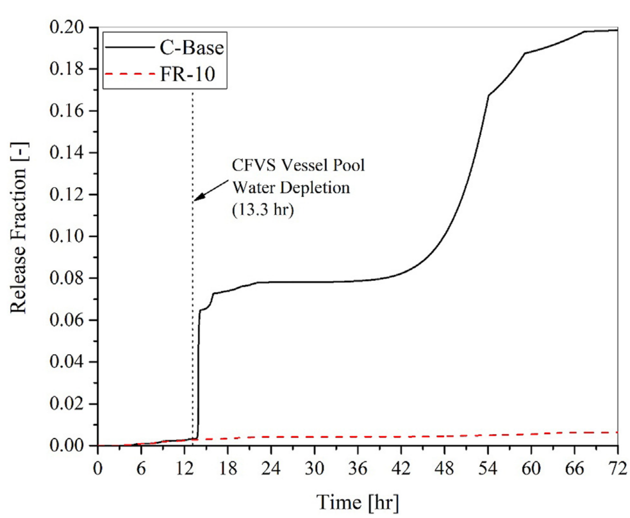

Fig. 10. Release fraction of Cs nuclide for C-Base and FR-10 cases.

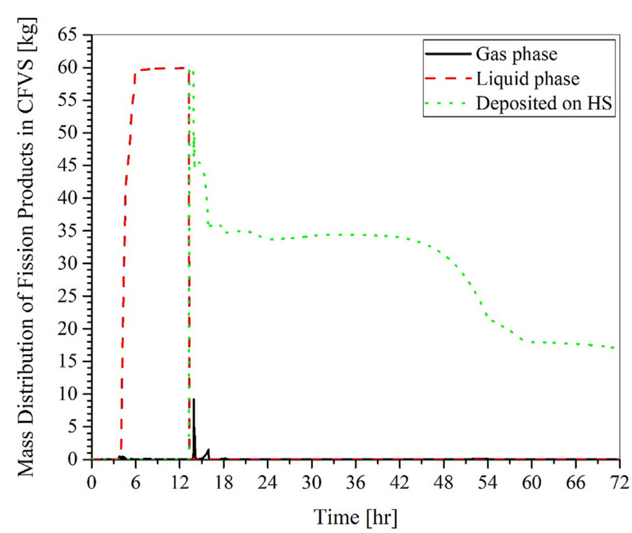

Fig. 12. Mass distribution of fission products in CFVS for C-Base case.

case for Cs nuclide.

Fig. 13 shows release fractions over time for Cs nuclide in the C-Base and FR- cases. The FR-cases that refill CFVS vessel with a certain flow rate show release fractions one or two orders of magnitude lower compared to the C-Base case. In other words, in order to maximize the efficiency of FP reduction using CFVS, continuous refill should be performed to avoid water depletion in the CFVS vessel.

5.3. *Effect of refill rate*

The FR- cases assume that the CFVS valve is operated 10 min into the SAMG entry, and the water level for starting refill is 1.5 m. Sensitivity analyses are conducted with varying refill rates of 0.1, 0.5, 1, and 10 kg/s and the results are shown in Table 13, Figs. 13 and 14.

Table 12
MELCOR simulation results for C-Base and FR-10 cases.

Accident Progress	MELCOR Simulation Results	
	C-Base	FR-10
Accident progression before operating CFVS valve is the same as the reference scenario (refer to Table 3)		
CFVS Valve Operation [sec]	12,306 (3.42 h)	
Cladding Failure [sec]	14,123 (3.92 h)	
RPV Failure [sec]	21,530 (5.98 h)	
Containment Pressure at 72 h [kPa(a)]	259	263
Cumulative Release Fraction of Cs up to 72 h [–]	0.1985	0.0063
Cumulative Release Fraction of I up to 72 h [–]	0.2641	0.0354
Cumulative Release Fraction of Te up to 72 h [–]	0.2168	0.0159

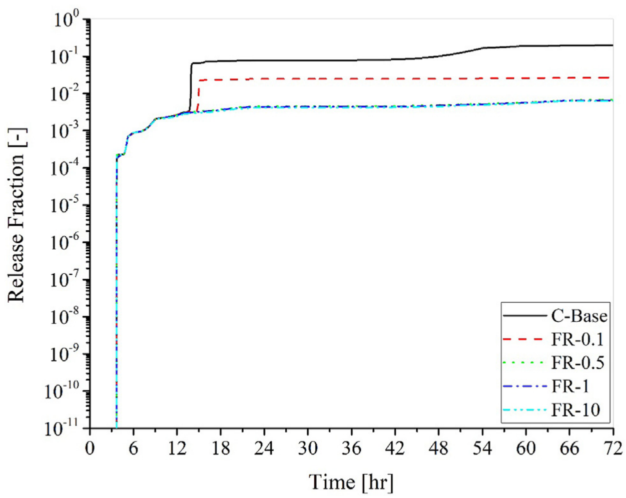

Fig. 13. Release Fraction of Cs nuclide for C-Base and FR- Cases.

As shown in the table, FP release fractions for FR-0.1 are larger than the other FR- cases, and it is because of the availability of the pool scrubbing. According to Fig. 14, refill rate of 0.5 kg/s maintains the CFVS water level above 1.5 m, keeping the sparger submerged all the time. In other words, if the minimum injection rate is available, the FP release fraction do not depend on the injection rate. When the CFVS pool scrubbing is working, only about 0.6% of the initial inventory of Cs, for example, is released to the environment.

5.4. *Effect of refill level*

The effect of FP reduction in conformance with the water level when to start injection into the CFVS vessel is analyzed. The WL-1.5 case means refill starts when the water level reaches 1.5 m. Sensitivity analyses are performed by changing the injection initiation level to 1.5, 2.0, 2.5, and 3.0 m. For the WL- cases, the CFVS valve is operated 10 min after the SAMG entry, and the refill rate is fixed at 10 kg/s. The flow rate of 10 kg/s is the case that can represent the normal operation of the CFVS with sufficient water injection.

Simulation results for the release fractions and water levels are shown in Table 14, Fig. 15 and Fig. 16, respectively. These results indicate that pool scrubbing is again an important contributor to control the FP release.

The results of section 5.3 and 5.4 are summarized as follows:

- Even a small amount of refill (0.5 kg/s) can maintain the water level of the CFVS vessel pool.
- CFVS refill before the sparger uncovery can enhance the FP reduction rate.

5.5. *Effect of CFVS valve opening time after SAMG entry*

Finally, sensitivity analysis on the timing of the CFVS valve operation is performed. The valve operation time cases are divided in the same manner as in section 4.1. For the T- cases, the refill rate is fixed at 10 kg/s after the water level is below 1.5 m.

As shown in Tables 15 and 16, and Fig. 17, the T-10, 20, and 30 cases have high FP reduction rates of more than 90%, while the T-40 and 50 cases experience relatively large decreases in reduction rate (55.9–70.9%). It is related to the time when cladding is damaged, and is due to the same mechanism as described in section 4.1. However, even if the total release fractions of T-20 and T-30 cases are lower than the C-Base, the initial release amounts are higher than the T-10 case due to the delay in CFVS valve operation. Therefore, it is important to operate the CFVS valve as early as possible after the SAMG entry. Table 16 summarizes the results of sensitivity analyses related to CFVS (see Fig. 18).

5.6. *Adverse effects*

Measures to reduce FP release using CFVS show little increase in containment pressure, unlike the RCS direct depressurization strategy proposed in the SAMG. As shown in Table 17, the highest containment pressure rise in the CFVS sensitivity cases is 5.1% by MCCI and containment heating, which is lower than that of the PORV cases that increases by 13.7–19.5%.

Although there is negligible adverse effect on increasing containment pressure, there is a possibility of hydrogen combustion and detonation in CFVS vessel and pipe during CFVS operation. Hydrogen generated during accident progression can flow into CFVS via the broken SG tube. In this case, structures such as CFVS vessel and pipe could be damaged, resulting in loss of filtering function.

Table 13
MELCOR simulation results for FR- cases.

Accident Progress	MELCOR Simulation Results			
	FR-0.1	FR-0.5	FR-1	FR-10
Accident progression before operating CFVS valve is the same as the reference scenario (refer to Table 3)				
CFVS Valve Operation [sec]	12,306 (3.42 h)			
Cladding Failure [sec]	14,123 (3.92 h)			
RPV Failure [sec]	21,530 (5.98 h)			
Containment Pressure at 72 h [kPa(a)]	259	261	262	263
Cumulative Release Fraction of Cs up to 72 h [–]	0.0269	0.0067	0.0066	0.0063
Cumulative Release Fraction of I up to 72 h [–]	0.0811	0.0361	0.0358	0.0354
Cumulative Release Fraction of Te up to 72 h [–]	0.0638	0.0169	0.0163	0.0159

Table 14
MELCOR simulation results for WL- cases.

Accident Progress	MELCOR Simulation Results			
	WL-1.5	WL-2.0	WL-2.5	WL-3.0
Accident progression before operating CFVS valve is the same as the reference scenario (refer to Table 3)				
CFVS Valve Operation [sec]	12,306 (3.42 h)			
Cladding Failure [sec]	14,123 (3.92 h)			
RPV Failure [sec]	21,530 (5.98 h)			21,540 (5.98 h)
Containment Pressure at 72 h [kPa(a)]	263		264	263
Cumulative Release Fraction of Cs up to 72 h [−]	0.0063	0.0064	0.0064	0.0064
Cumulative Release Fraction of I up to 72 h [−]	0.0354	0.0355	0.0353	0.0353
Cumulative Release Fraction of Te up to 72 h [−]	0.0159	0.0158	0.0157	0.0156

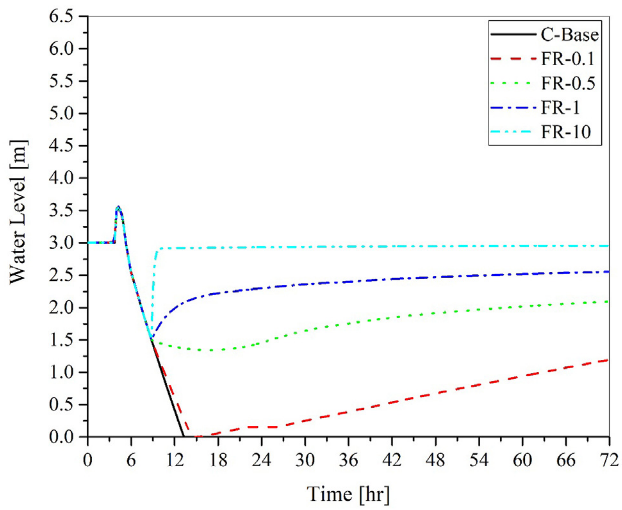

Fig. 14. Water level in CFVS vessel for C-Base and FR- cases.

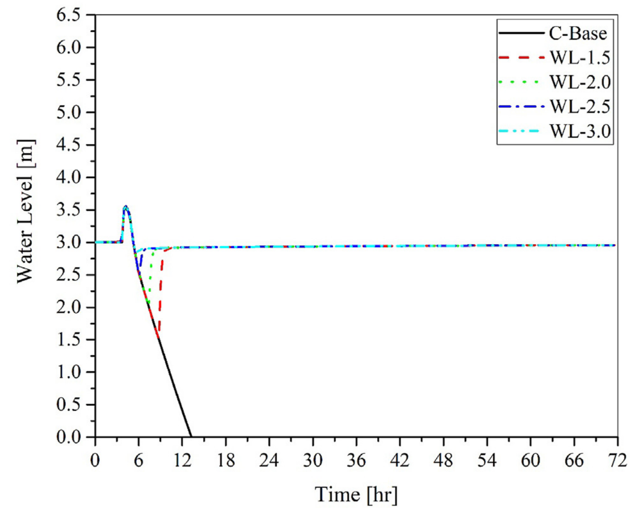

Fig. 16. Water level in CFVS vessel for C-Base and WL- cases.

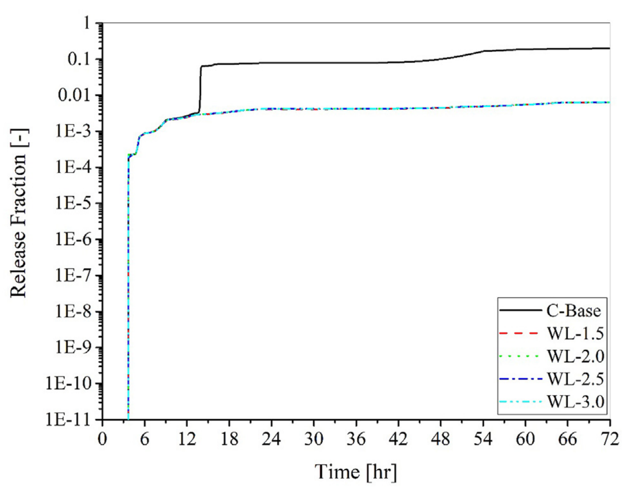

Fig. 15. Release Fraction of Cs nuclide for C-Base and WL- Cases.

In this section, the possibility of hydrogen combustion and detonation in the CFVS vessel is analyzed using the Shapiro diagram (refer to Fig. 19) [17]. The FR-10 case is utilized to analyze hydrogen risk. The concentration of hydrogen and steam in the CFVS vessel begins to rise, entering the combustion zone at 13,420 s (3.73 h). Hydrogen concentration continues to rise, reaching a detonation limit at 13,530 s (3.76 h), and leaving the zone at 13,710 s (3.81 h). The concentration exceeds 53% at 13,790 s (3.83 h), leaving the combustion zone. Afterward, hydrogen is gradually removed along pipes from the top of CFVS to the environment. Hydrogen concentration increases again later in the accident, but hydrogen does not enter the combustion zone again.

In order to reduce the hydrogen risk, an igniter needs to be installed. When the igniter operates inside the CFVS vessel, oxygen and hydrogen in the atmosphere in the vessel are burned and do not enter the detonation zone (refer to Fig. 19). However, further research is needed to reduce risk by controlling hydrogen concentration. In addition, hydrogen risk analysis also requires a detailed analysis of temperature, pressure and shape in CFVS vessel, and the uncertainty assessment.

6. Conclusions

In this paper, we propose an introduction of installing three-way valves downstream of SG PRV and MSSV to deliver the steam containing FPs in the main steam pipe into the CFVS vessel. It can reduce a large amount of FP release while minimizing adverse effects identified in the previous studies. In order to compare the effect of new equipment in conformance with the existing strategy, MELCOR simulations have been performed for the series of accidents. As a result of simulations, it is confirmed that CFVS operation lowers the FP release into the environment, and the release fractions are lower (minimum 0.6% of the initial inventory for Cs) than that of the strategy which depressurizes the primary system

Table 15
MELCOR simulation results for T- cases.

Accident Progress	MELCOR Simulation Results				
	T-10	T-20	T-30	T-40	T-50
Accident progression before operating CFVS valve is the same as the reference scenario (refer to Table 3)					
CFVS Valve Operation [sec]	12,306 (3.42 h)	12,906 (3.59 h)	13,506 (3.75 h)	14,106 (3.92 h)	14,706 (4.09 h)
Cladding Failure [sec]	14,123 (3.92 h)				
RPV Failure [sec]	21,530 (5.98 h)	21,284 (5.91 h)	21,093 (5.86 h)	21,816 (6.06 h)	20,682 (5.75 h)
Containment Pressure at 72 h [kPa(a)]	263	262	266	262	266
Cumulative Release Fraction of Cs up to 72 h [−]	0.0063	0.0140	0.0260	0.1261	0.1909
Cumulative Release Fraction of I up to 72 h [−]	0.0354	0.0377	0.0483	0.1983	0.3163
Cumulative Release Fraction of Te up to 72 h [−]	0.0159	0.0152	0.0232	0.1509	0.2588

Table 17
Pressure rise of containment building compared to reference scenario (CFVS sensitivity cases).

Cases	Containment Pressure at 72 h [kPa(a)]	Pressure Rise Rate of Containment Building [%]
C-Base	263	2.3
FR-0.1	259	2.3
FR-0.5	261	3.1
FR-1	262	3.5
FR-10	263	3.9
WL-1.5	263	3.9
WL-2.0	263	3.9
WL-2.5	264	4.3
WL-3.0	263	3.9
T-10	263	3.9
T-20	262	3.5
T-30	266	5.1
T-40	262	3.5
T-50	266	5.1

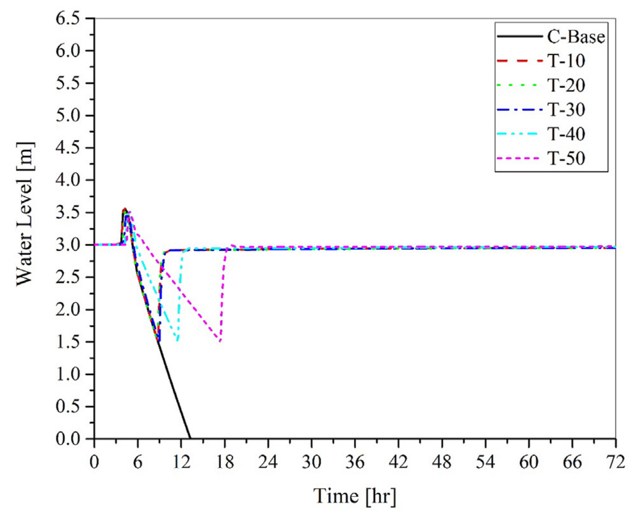

Fig. 18. Water level in CFVS vessel for C-Base and T- cases.

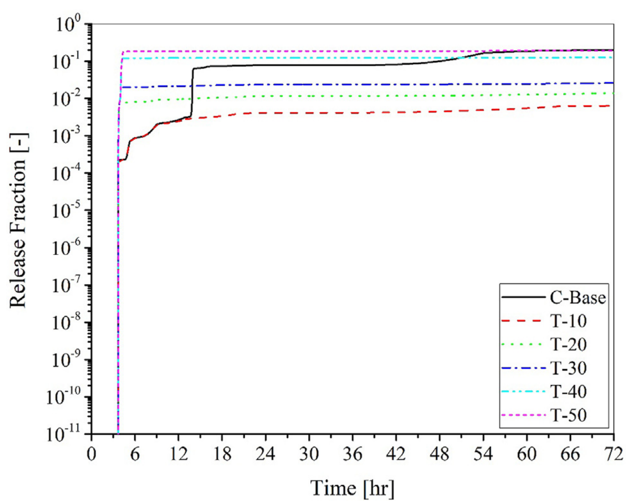

Fig. 17. Release Fraction of Cs nuclide for C-Base and T- Cases.

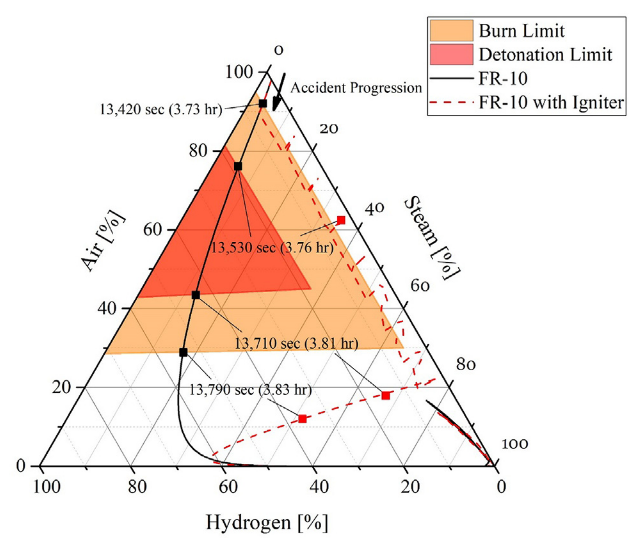

Fig. 19. Shapiro diagram for hydrogen-air-steam mixtures for FR-10 case.

directly (minimum 15.2% for Cs). By conducting sensitivity analyses, it is identified that the refill of the CFVS vessel is a dominant contributor to controlling the amount of FP release to the environment. Hence, the release fraction of Cs to the environment can be reduced to 0.6% of the initial inventory for the case of refill the CFVS vessel optimally (FR-10), compared to 19.8% for no refill case (C-Base). The new strategy using CFVS has no adverse effects on increasing containment pressure. However, it raises a possibility of hydrogen combustion and detonation in the CFVS vessel. Installation of an igniter inside the CFVS vessel could be considered to reduce the hydrogen risk, but a more detailed analysis is needed.

The results of this study can be used as a basis for further research on off-site risk reduction through FP release control, and be used to identify optimal implementation methods supporting severe accident management strategy. The new measure is also expected to be adopted complying with an accident management strategy to meet the recently announced NPP safety targets in Korea.

Declaration of competing interest

The authors declare that they have no known competing financial interests or personal relationships that could have appeared to influence the work reported in this paper.

Acknowledgments

This work was supported by the Nuclear Safety Research Program through the Korea Foundation Of Nuclear Safety (KOFONS), granted financial resource from the Multi-Unit Risk Research Group (MURRG), Republic of Korea (No. 1705001).

Appendix A. Supplementary data

Supplementary data to this article can be found online at https://doi.org/10.1016/j.net.2020.08.003.

References

[1] S.A. Eide, T. Weirman, D. Gentillon, D. Rasmuson, C. Atwood, NUREC/CR-6928: Industry-Average Performance for Components and Initiating Events at US Commercial Nuclear Power Plants, US Nuclear Regulatory Commission, 2007, p. 42. NUREG/CR-6928.
[2] Nuclear Safety and Security Commission, Regulation on the Scope of Accident Management and the Detailed Criteria for Evaluating Accident Management Capabilities, 2017. Notice No. 2017-2034.
[3] R.J. Lutz, Westinghouse Owners Group Severe Accident Management Guidance Validation, Westinghouse Electric Corporation, Nuclear Technology Division, 1994.
[4] W. Choi, H-Y. Kim, R-J. Park, S. J. Kim, Effectiveness and adverse effects of in-vessel retention strategies under a postulated SGTR accident of an OPR1000, J. Nucl. Sci. Technol., https://doi.org/10.1080/00223131.2016.1273145.
[5] T. Kim, W. Choi, J. Jeon, N. K. Kim, H. Jung, S. J. Kim, A conceptual approach to eliminate bypass release of fission products by in-containment Relief valve under SGTR accident, Sci. Technol. Nucl. Ins., https://doi.org/10.1155/2018/5936214.
[6] Korea Hydro & Nuclear Power Co, Ltd, Final Safety Analysis Report for Kori Unit, vol. 2, 1989.
[7] L.L. Humphries, B.A. Beeny, F. Gelbard, D.L. Louie, J. Philips, MELCOR Computer Code Manuals Vol. 1: Primer and Users' Guide Version 2.2.14959, Sandia National Laboratories, Albuquerque, 2019. SAND2019-12536 O.
[8] Us Nuclear Regulatory Commission, State-of-the-Art Reactor Consequence Analyses Project, vol. 2, Surry Integrated Analysis, 2012. NUREG/CR-7110, vol. 2.
[9] Korea Hydro & Nuclear Power Co, Ltd, Severe Accident Management Guidance: Mitigation-02, Reactor Coolant System Depressurization, 2015.
[10] Korea Hydro & Nuclear Power Co, Ltd, Probabilistic Safety Assessment Report for Kori Unit 2: Containment Integrity Analysis, 2015.
[11] S.H. Lee, Containment building analysis considering containment filtered venting system (CFVS), in: 2014 Nuclear Safety Analysis Symposium, Daejeon, Korea, 2014. July 17.
[12] Committee on the Safety of Nuclear Installations, Status Report on Filtered Containment Venting, NEA/CSNI/R(2014)7, 2014.
[13] M. Loffler, M. Braun, Evaluation of filtered containment venting systems with MELCOR for extended operating conditions in German PWR, in: European MELCOR and MACCS User Group Meeting, Bratislava, Slovakia, 2014. April 15-16.
[14] M. Welker, AREVA's containment venting technologies and experience worldwide, in: Sociedad Nuclear Espanola, A Coruna, 2015. Spain, Sepp. 23–25.
[15] Y. S. Na, K. S. Ha, R-J. Park, J-H Park, S-W. C, Thermal hydraulic issues of containment filtered venting system for a long operating time, Nucl. Eng. Technol., https://doi.org/10.5516/NET.02.2014.031.
[16] L.L. Humphries, B.A. Beeny, C. Faucett, F. Gelbard, T. Haskin, D.L. Louie, J. Phillips, MELCOR Computer Code Manuals Vol. 2: Reference Manual Version 2.2.14959, Sandia National Laboratories, Albuquerque, 2019. SAND2019-12537 O.
[17] Y. S. Na, S-W. Cho, K. S. Ha, The hydrogen issue in the initial operation of a filtered containment venting system, Nucl. Technol.., https://doi.org/10.13182/NT15-160.