DEVELOPMENT AND APPLICATION OF SOURCE TERM ANALYSIS FRAMEWORK ON OPR-1000 NPP FOR RADIOLOGICAL EMERGENCY PREPAREDNESS

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For the radiological emergency preparedness of nuclear power plant, it is needed to improve the understanding on the thermal hydraulic behaviors and fission product behaviors when severe accident occurs in the nuclear power plant. For the establishment of correct and effective emergency preparedness, source term characterization is very important. To establish the source term analysis framework, the following works are performed in this project.

MELCOR analyses are performed on the SBO (Station Blackout) sequence with containment over-pressurization failure and an SGTR (Steam Generator Tube Rupture) sequence which is a typical containment bypass scenario. Source term analysis on the above accident scenarios, that is, 1) release of radioactive fission products from the fuel, 2) transport and deposition characteristics in reactor vessels and containment structures, and 3) the release timing, release duration and release magnitude of 12 radio nuclide classes to the environment.

A trial application on the development of source term release characteristic map to environment are performed for the above-mentioned accident scenarios in view of the release timing, release duration and release magnitude of RN classes to the environment and deposition characteristic of radioactive fission products in reactor coolant system and containment. Resulting environmental release fractions are compared with that of Fukushima Daiichi accident. Release charateristics to environment between volatile and non-valitile radionuclides will be presented between different reactor types (PWR vs BWR) and different accident sequences (SBO vs SGTR) in the following sections.

I. INTRODUCTION

Since the Three Mile Island (TMI) (1979), Chernobyl (1986), Fukushima Daiichi (March 11, 2011) accidents, the assessment of radiological source term effects on the environment has been a key concern of nuclear safety. There is a long history of applying radiological source terms to the reactor risk study, siting criteria development and radiological emergency preparedness of the light water reactors in USA: TID-14844, NUREG-1465 (Accident Source Terms), WASH-1400, NUREG-1150, etc. Recently, the SOARCA project (US NRC, 2012) in U.S. NRC (Nuclear Regulation Commission) has treated long-term and short-term SBO accident sequences for Surry (Large dry containment PWR) and Peach Bottom (MARK I BWR) plants and presented the reduced release amounts of radiological source term with the current-state-of-the-art knowledge of radiological transport in the severe accident environment by MELCOR code (US NRC, 2005). Since the Fukushima Daiichi accident, Japanese government and international organizations such as IAEA, UNSCEAR, WHO published on the impact of fission product release to the environment.

Mark I BWR is installed at Fukushima Daiichi Units 1, 2, and 3. In Fukushima accident, off-site and on-site AC (alternate current) powers were lost by tsunami attack about 45 minutes after earthquake. Loss of Isolation Cooling (IC) occurs also in this time in Unit 1. DC (direct current) battery power was immediately lost in Unit 1 by the tsunami attack. Even though we don't know the exact time when the DC battery powers were lost in Units 2 and 3, it is known that the cooling function operated by reactor core isolation cooling/ high pressure core injection (RCIC/HPCI) were lost about 72 and 36 hours after the tsunami attack in Units 2 and 3, respectively. Over-pressurization in cores due to the loss of cooling functions make SRV in RPV opening to PCV. Steam, hydrogen, and fission product aerosols generated in core is released to the PCV by opening of SRV. High pressure (more than 0.75 MPa) built up in PCV make release or leakage path to reactor building or turbine building. Radiological fission products are started to release to atmosphere approximately at 20 (Unit 1), 40 (Unit 3), and 80 h (Unit 2) after the earthquake. The off-site AC power was recovered in 9 days after the accident in the nuclear power station (NPS) site. Therefore, external cooling water injection by fire pump truck with fresh water or seawater is only available in the

Fukushima accident before the recovery of off-site power. However, cooling water injection by fire pump truck is not always effective due to the high pressure of RPV inside or leakages/bypasses of external water injection flow paths. SRV opening due to high RPV pressure make a release path of hydrogen, steam, and fission product aerosols to PCV. Loss of coolant to RPV makes reactor core fuel heat up and makes RPV lower head rupture subsequently. High pressure built up in PCV makes drywell head flange rupture. Released hydrogen makes reactor building explosion in Units 1 and 3 or makes the blowout panel rupture in Unit 2. These ruptures of buildings and venting trials in suppression chamber and drywell give results in massive fission products release to atmosphere.

Likewise, sustained SBO in PWR make the RPV /RCS heated up and PZR PORV should be open to release of pressure built up RPV due to the loss of cooling capability in primary and secondary sides of steam generator. If PORV opens appropriately, then the energy generated in primary side will be decipated to containment through PROV opening. The released energy in containment and MCCI phenomena in containment cavity will make containment pressure build up subsequently. Finally, containment pressure will be reached ultimate failure pressure. If PORV is not open appropriately then pressurizer surge line or steam generator U-tubes will be failed due to the temperature stress of pipings. Steam generator tube rupture or interfacing loss of coolant accident is typical containment bypass scenario in PWR. Fission product generated in core will be released to the environment directly without depositing inside of the containment. These sequences are called containment bypass sequences. No external cooling water injection is assumed for both SBO and SGTR cases in PWR severe accident scenarios.

2. MELCOR MODELING FOR OPR-1000

Temperatures of cladding and fuel nodes are calculated by COR Package of MELCOR code. If the temperature is less than 1173 K (900°C) for any node, no release will occur from that node. The temperature for failure of the cladding of a fuel rod is taken to be 900°C.

The reference plant adopted is an Optimized Power Reactor (OPR-1000) type plant, which is typical of Korean plants (http://www.opr1000.co.kr/). These plants are two-loop (2 steam generator) type PWR with a 2815MW thermal power and housing a large dry containment. Thermal-hydraulic (CVH package) and flow-path (FL package) nodalization in MELOCR for the RCS (reactor coolant system) and containment of reference plant are shown in Fig. 1 and 2 respectively.

The elevations of control volumes are set from reference level of hot leg centerline (0.0 m). The total free volume of RCS except pressurizer is about 290 m^3. Lower Plenum, Core, and Bypass control volumes are linked with COR (core) package. COR cells consist of 13 levels and 7 radial rings. Core materials which can be molten during severe accident scenarios are 86 tons of fuel, 24 tons of Zircaloy cladding, and 12 tons of core supporting structural material of stainless steel (tables 1 and 2). Each core cell has their initial fission product inventory. When the core is heated up and relocated to lower regions, decay heat, fission product inventory is moved according to the movement of molten core or core debris. Fission product aerosols are transported according to the movement of steam in RCS and containment. When RPV lower head failure occurs, the molten corium or particulate core debris will be relocated to the reactor cavity below the RPV. Hot corium ejected to the cavity (temperature is about 2500K) have interaction with concrete on the basement floor in cavity. This is called molten core and concrete interaction (MCCI) phenomena. Non condensable gases such as CO, CO2, H2, H2O will be generated during MCCI in cavity and these non-condensable gases will make pressure build up in containment. If pressure reached containment failure pressure then fission product aerosols will be escaped from containment to the environment. Containment failure area and failure pressure are assumed by the user. Intact Zircaloy and stainless steel which are not yet oxidized in RPV will be oxidized in cavity again. There are typically two kinds of concretes; one is calcious concrete and the other is silicious concrete. Non condensable gas generation rate, mixing rate, and concrete ablation rate (axial and radial) are changed depending on the concrete properties and corium properties and mass, etc. Flat-bottom cylindrical geometry is usually assumed for the shape of cavity concrete.

When the SBO accident occurs in the PWR plant, only turbine driven auxiliary feedwater (TD-AFW) pump can be available if vital DC battery power is available. However, TD-AFW pump is also assumed to be not working with the assumption of DC power loss. In this worst condition, there is only one way to relieve the pressure of RCS by cyclic opening of PORV (pilot operated relief valve). During this sequence, RCS inventory is released to pressurizer relief tank. The volume of PRT (pressurizer relief tank) is not large enough so the rupture disc will open eventually. The water and steam will be released to the containment atmosphere. The pressure in the containment builds up and will reach containment failure pressure. Four passive safety injection tanks (SIT) with total 200 tons of water are available. The water is automatically injected into the RCS when RCS pressure drops below 4 MPa. No external cooling water injection is assumed into primary and secondary sides of steam generator for both SBO and SGTR cases in PWR severe accident scenarios

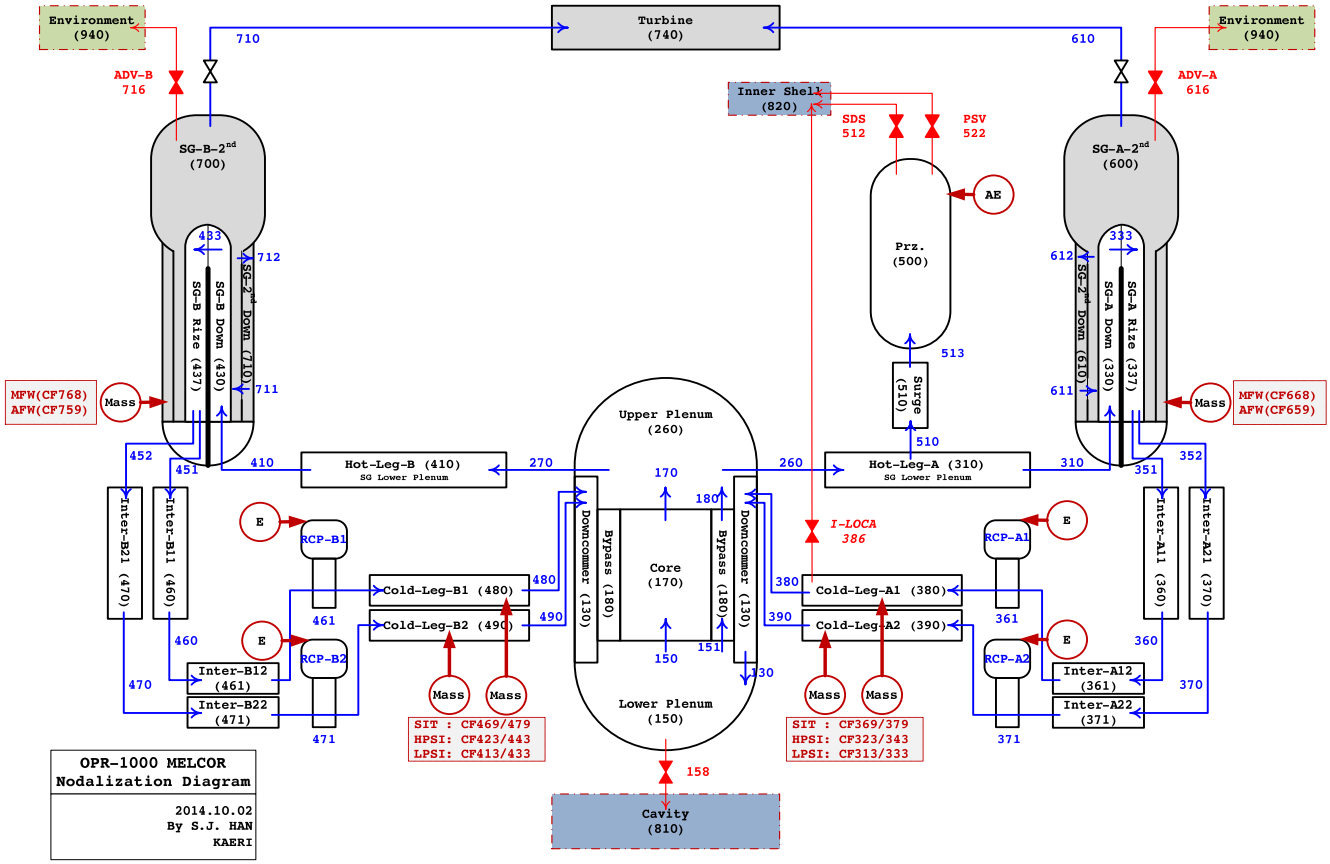

Fig. 1 MELCOR CVH/FL nodalization diagram for OPR-1000, typical Korean PWR

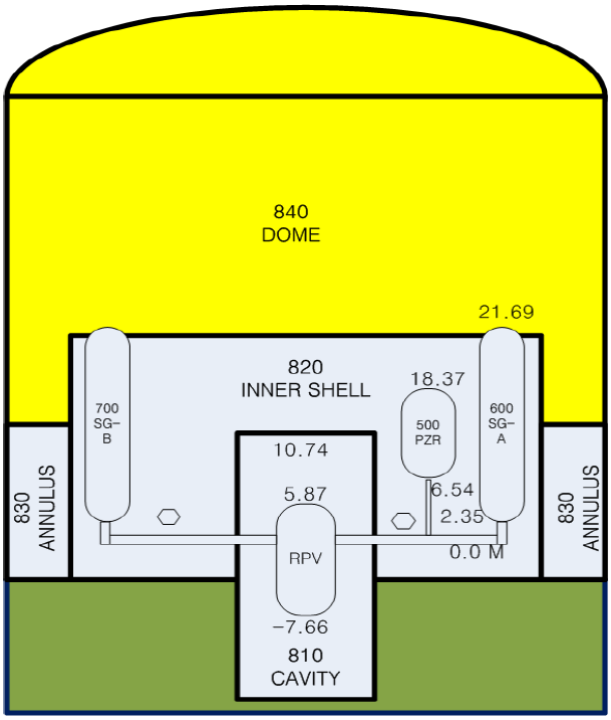

Fig.2 MELCOR Nodalization for OPR-1000 Containment

Table 1. Initial Conditions of Plant

Plant Parameters	Value
Nominal Reactor Power (MWth)	2815
Decay Heat When Reactor Trip Occurs (6% of Nominal) (MWth)	23.9
Initial RCS Free Volume excluding Pressurizer Volume (m3)	288
Four SITs Total Water Inventory (tons)	200

Table 2. Initial Mass of Core Materials

Core Material	Mass (tons)
UO2 Fuel	85.6
Zircaloy	23.9
Stainless Steel	11.7
Total	121.2

Source Term Estimation Methodology in MELCOR code

MELCOR code version 1.8.6 is used in this analysis. Thermal hydraulic behaviors are estimated by using control volume and flow path approach in MELCOR code. Core heatup and relocation calculations are performed after coolant inventory loss. Initial fission product inventory in fuel matrix is usually estimated by ORIGEN code. When the fuel cladding temperature reaches above a certain temperature, cladding oxidation reaction occurs. If cladding oxidation reaction occurs, fission products, which were resided in fuel will be released to reactor core and coolant regions. Thereafter, fission product release from fuel, transport inside RCS and finally moving to containment is estimated. Fission product aerosol dynamics are used to describe the movement of fission product inside the plant. Fission product generation and deposition on structures in each control volume are estimated. When the reactor vessel lower head failure occurs, corium movement to reactor cavity is estimated. The corium relocated to cavity interacts with concrete, which resides in cavity bottom floor.

The release rate of fission products from fuel matrix is calculated by CORSOR model in MELCOR code. Total 16 radionuclide (RN) classes are treated in MELCOR code as shown in Table 3. Each RN class has different release rate from core according to the heat up of fuel. Volatile fission product, such as noble gases (Xe, Kr), Cs, I, and Te, released out from fuel more rapidly than other less volatile or non-volatile radionuclide. In MELCOR code version 1.8.6, there are 16 aerosol classes treated, which is shown in Table 4. Class 15 “concrete” is defined for treating MCCI in cavity. Class 16 CsI is defined for treating CsI, which is assumed to be generated when I is released from fuel to RCS and combined with Cs immediately. Transport and deposition rates of fission products in reactor coolant system (RCS) and containment structures are estimated by MAEROS module. MAEROS module treat fission products agglomeration and deposition behaviors with help of multi-sectional (size bins) aerosol dynamics. The pressure and temperature in the control volumes affect to the aerosol dynamics. Release characteristics between volatile and non-volatile radionuclides will be presented between different reactor types (PWR vs BWR)/ different accident sequences (SBO vs SGTR) in the following sections.

In MELCOR code, RN (Radionuclide) Package handles volatile fission products release from fuel pellet to core coolant, transport and deposition of aerosols through RCS, and movement of non-volatile fission products to reactor cavity when lower head failure occurs and finally movement of radioactive and non-radioactive materials to the environment through containment failure openings. In the containment bypass sequences such as SGTR and ISLOCA, fission products can be released directly to the environment through SG secondary side or low pressure RCS boundaries such as residual heat removal system (RHRS), shutdown cooling system (SCS), and low pressure safety injection (LPSI) system. Volatile radioactive aerosols are entrained by the steam. Non-volatile fission products move by being contained in the fuel debris beds or molten corium.

In each control volume, MAEROS module is used to calculate the aerosol size distribution. MAEROS is a multi-sectional, multi-component aerosol dynamics code that evaluates the size distribution of each type of aerosol mass, or component, as a function of time. This size distribution is described by the mass in each size bin, or section. Aerosols can directly deposit onto heat structure and water pool surfaces through four processes calculated within MAEROS. All heat structure surfaces are automatically designated as deposition surfaces for aerosols using information from the HS package, unless made inactive through user input.

The MAEROS deposition kernel for each type of surface is made up of four contributions: gravitational deposition, Brownian diffusion to surfaces, thermophoresis, and diffusiophoresis. Of these natural depletion processes, gravitational deposition is often the dominant mechanism for large control volumes such as those typically used to simulate the containment, although diffusiophoretic effects may be significant in some cases (e.g., diffusiophoresis during water condensation). Particle diffusion

is generally considered to be a relatively unimportant deposition process.

Table 3. MELCOR RN Class Compositions

Class Name	Representative	Member Elements
1. Noble Gases	Xe	He, Ne, Ar, Kr, Xe, Rn, H, N
2. Alkali Metals	Cs	Li, Na, K, Rb, Cs, Fr, Cu
3. Alkaline Earths	Ba	Be, Mg, Ca, Sr, Ba, Ra, Es, Fm
4. Halogens	I	F, Cl, Br, I, At
5. Chalcogens	Te	O, S, Se, Te, Po
6. Platinoids	Ru	Ru, Rh, Pd, Re, Os, Ir, Pt, Au, Ni
7. Early Transition Elements	Mo	V, Cr, Fe, Co, Mn, Nb, Mo, Tc, Ta, W
8. Tetravalent	Ce	Ti, Zr, Hf, Ce, Th, Pa, Np, Pu, C
9. Trivalents	La	Al, Sc, Y, La, Ac, Pr, Nd, Pm, Sm, Eu, Gd, Tb, Dy, Ho, Er, Tm, Yb, Lu, Am, Cm, Bk, Cf
10. Uranium	U	U
11. More Volatile Main Group	Cd	Cd, Hg, Zn, As, Sb, Pb, Tl, Bi
12. Less Volatile Main Group	Sn	Ga, Ge, In, Sn, Ag
13. Boron	B	B, Si, P
14. Water	H_2O	H_2O
15. Concrete	- -	- -
16. Cesium Iodide	CsI	Classes 2 and 4

3. THERMAL-HYDRAULIC RESULTS

3.1 SBO Sequence

Here the worst situation is assumed such as the long-term loss of on-site and off-site AC powers for more than a few days duration that engineered safety features such as safety injection pumps and motor-driven auxiliary feedwater (MD-AFW) pumps cannot work during this time period. In the SBO situations in pressurized water reactor plant (PWR), turbine driven auxiliary feedwater (TD-AFW) pump can inject water to the secondary side of steam generator. However, turbine inlet steam flow control valve cannot work properly when loss of vital DC power occurs. Vital DC power is designed to be maintained during 4 or 8 hours in the SBO conditions. In this paper motor-driven and turbine driven AFW pumps are all assumed to be not working at time 0 sec as a worst case assumption. It is necessary to study a more detailed SBO considering its importance in the consequential effects, but there are a few of knowledge bases of radiological source term behaviors during long-term SBO accident.

Table 4 shows key events of given scenario. Top of active fuel (TAF) uncovers at 2.06 h. As core heats up, radioactive fission products, which were residing in fuel matrix or in fuel cladding gap region starts to release to reactor core channel (2.35 h) when the fuel and cladding temperature increase over about 1000K. Core degradation and relocation occur during from 2.35 to 4.16 h. Finally failures of reactor vessel lower head penetrations occur during about 1 hour from 4.16 h to 5.1 h. After lower head failure, MCCI (molten corium concrete interaction) occurs in the cavity below the reactor vessel. During the MCCI process in the reactor cavity, non-condensable gases such as CO_2, CO, H_2, and H_2O are generated and these non-condensable gases increase the containment pressure. When containment pressure reaches 7 MPa, it is assumed that containment failure occurs. Fission product aerosols were released from fuel and transported and deposited on the walls of RCS piping and containment structures. When containment failure or leakage occurs, fission products aerosols in the containment atmosphere release to environment.

Fig.3-1A shows RCS pressure transients. RCS pressure drops to 0.4 MPa and containment pressure increases to 0.4 MPa at 4 h due to the failure of lower head. Containment pressure spike at 10 h might happen due to the hydrogen deflagration. The

amount of hydrogen generated in the cavity due to MCCI is larger than the amount of hydrogen generated in RPV due to the metal-water reaction (MWR). Containment failure occurs at 43 h due to containment pressure reaches 0.7 MPa. It is assumed that containment failure occurs at 0.7 MPa.

Fig.3-1C shows decay heat distribution in core and cavity. In Fig.3-1C, the difference of power between core decay power and actual core power rate between 2.5 to 4.2 h represent the transport amount of volatile radiological species (such as noble gases, iodine, cesium, tellurium) leaving the core region to other compartments (for instance, to the upper plenum and to the two RCS loops). After the vessel breach at 4 h, most of the molten core materials are ejected to the cavity between 4 to 5 h. At about 2.6 and 2.8 h, 200 and 100 MW of heat are generated from metal water reaction, which are much greater than whole core decay heat of 30 MW at this time frame.

Fig.3-1D shows key features of the core degradation. Core degradation materials consist of UO_2, Zircaloy, Stainless Steel, etc. and relevant oxidation increases according to severe accident progression. Zircaloy is changed to ZrO_2 at 2.5 h. Core materials ejected to cavity from 4 h to 5 h. Core region is modeled as 7 concentric radial rings.

Fission products deposited on the walls of RPV and RCS will be escaped first to the containment and then released to the environment eventually by the evaporation process due to the pressure difference between the RCS and containment (0.7 MPa) and the environment (0.1 MPa).

Fig.3-1E shows non condensable gas generations during MCCI process in cavity. You can see that CO gas generation rate is decreased and CO2 gas generation rate is increased about 15 h. You can see also that H2 generation rate is decreased and H2O generation rate is increased sharply at 15 h. This is because the zircaloy and steel not oxidized at RPV are react with concrete at the cavity from 4 h to 15 h. However, zircaloy and steel consumtion are exhausted at 17. Thereafter, CO2 and H2O generation rate is increased sharply after this time. At 43 h, which is the time of containment failure occurring, the noncondensable gas generation rates (slopes) are changed again.

3.2 SGTR Sequence

The typical Korean PSA report (KEPRI, 2002) denoted that SGTR has been nominated as the most hazardous severe accident scenario, because this accident makes a direct release path of radiological source terms of reactor core inventories. Under this background for SGTR simulation, a worst-case scenario of SGTR event was simulated by MELCOR version 1.8.6.

SGTR transients can vary by break size, availability of safety features, and operator actions. Because the present study is to derive basic features of radiological source term behaviors, a worst and critical scenario was selected: Double ended guillotine break of one tube size of SGTR accident is occurred at SG-A at time 0 s. At the same time with SGTR occurrence, the reactor is tripped at 0 sec with RCP trip, secondary steam line isolation (MSIV closure), main feedwater (MFW) stopped. All the active safety systems such as high pressure and low pressure safety injection systems (HPSI and LPSI) and motor-driven auxiliary feedwater (MD-AFW) pump are also not working due to the assumption of long-term SBO occurrence. Only turbine driven auxiliary feedwater (TD-AFW) pump can be available if vital DC battery power is available. However, TD-AFW pump is also assumed to be not working with the assumption of DC power loss. In this worst condition, there is only one way to release the pressure of SG secondary side. That is the opening of atmospheric dump valves (ADVs) installed in main steam line. During the automatic SG pressure release process, it is assumed that one ADV is stuck open at the timing of 5% equivalent area opening of one ADV full area at time of 10 s.

Table 5 shows key events of given scenario. Core uncovery (TAF) occurs at 3.25 h. As core heatup occurs, radioactive materials residing in fuel cladding gap region starts to release to reactor core channel (3.78 hr). Core degradation and relocation occur during from 4 to 10 h. Finally, failures of lower head penetrations occur from 10.3 h to 11.5 h.

Fig.3-2A shows RCS pressure transients. SG-A secondary side pressure starts to decease at about 4 h. RCS pressure follows SG-A pressure. RCS pressure drops to containment pressure at 10.3 h due to the failure of lower head.

Fig.3-2B shows RPV water level transients. Core uncovery (TAF) occurs at 3.25 h due to the coolant discharge to SG-A secondary side and to the environment finally. Reactor vessel water level reaches BAF (Bottom of active fuel) level reaches at 4 h. Core materials slumping down to lower plenum from 6 h to 10.3 h.

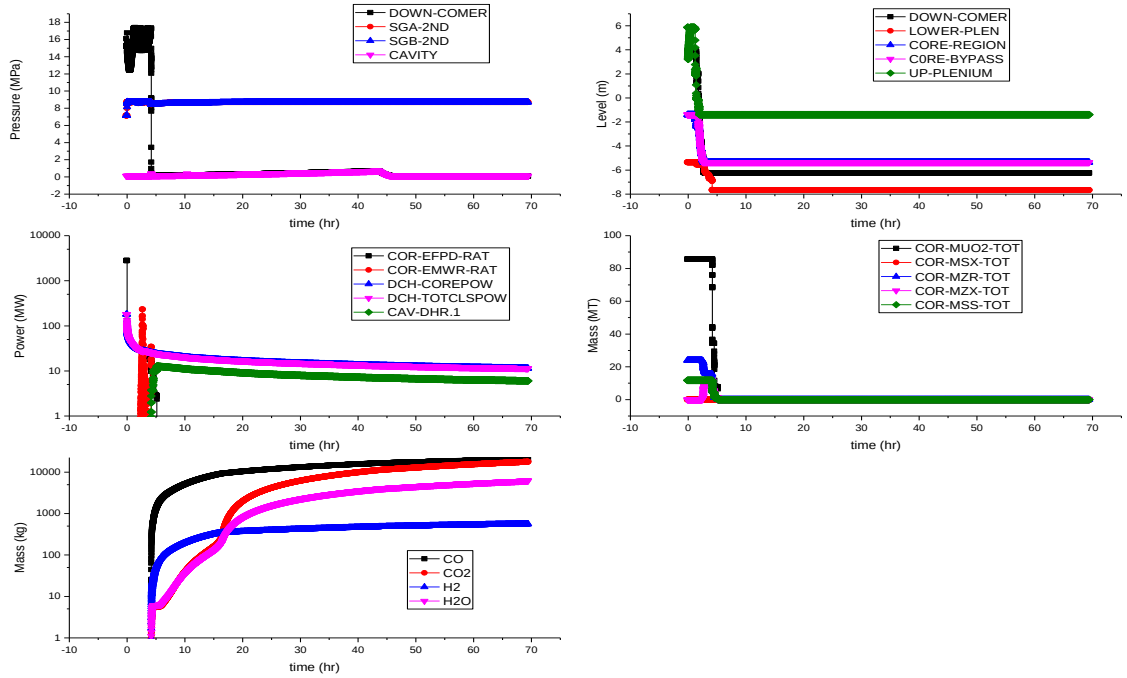

Fig.3-1 Thermal hydraulic behaviors in SBO case

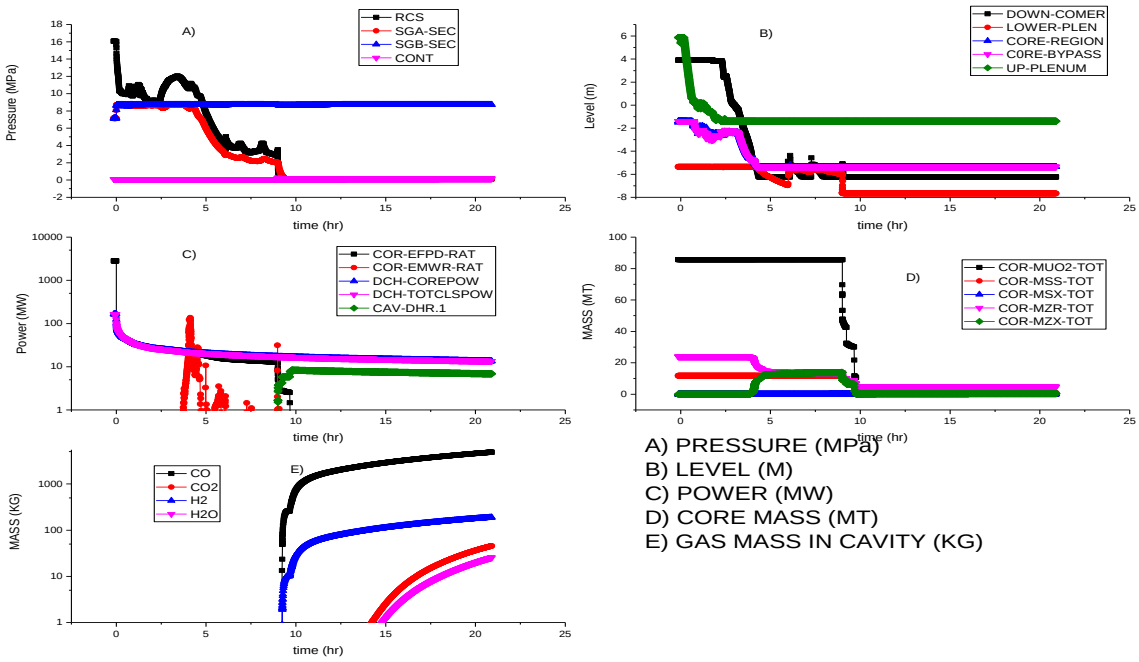

Fig.3-2 Thermal hydraulic behaviors in SGTR case

Fig.3-2C shows decay heat distribution in core and cavity. In Fig.9, the difference of power between core decay power (DCH-COREPOW) and actual core power rate (COR-EFPD-RAT) represents the transport amount of radiological materials leaving the core region to other compartments. After the vessel breach at 10.3 h, most of the molten core materials are ejected to the cavity and these ejected materials are involved in molten-core-concrete-interaction (MCCI) in the cavity during a period of 10.3 h to 11.5 h, which contributes long term behaviors of source term but adds limited amounts comparing with the core degradation period in the current SGTR case.

Fig.3-2D shows key features of the core degradation. Core degradation materials consist of UO2, Zircaloy, Stainless Steel, etc. and relevant oxidation increases according to severe accident progression. Zircaloy is changed to ZrO2 at 4 h. Core materials ejected to cavity from 10.3 h to 11.4 h. Core region is modeled as 7 concentric radial rings.

Table 4. Accident Progression of SBO Scenario

event	time (hr)
SBO occurs, Reactor trip, MFW trip, MSIV closure, AFW trip	0.0
SG-1, SG-2 dryout	1.6
Core uncover starts	2.1
Start of fission products release from fuel	2.4
Core support plate failure	3.8
RPV lower head (penetration) failure.	4.2
Types of gases generation during MCCI in cavity are changed H2 ➔ H2O, CO ➔ CO2	15
Containment Failure (P > 7 MPa)	43
Simulation ending time	70

Table 5. Accident Progression of SGTR Scenario

event	time (hr)
SG-A SGTR occurs, Reactor trip, MFW trip, MSIV close	0 s
SG-A ADV open	10 s
Core uncovery (TAF) occurs	3.3
Start of fission products release from fuel	3.8
Core support plate (CSP) failure starts	10.2
RPV lower head (penetration) failure.	10.3
Types of gases generation during MCCI in cavity are changed H2 ➔ H2O, CO ➔ CO2	14
Containment Failure (P > 7 MPa)	-
Simulation ending time	25

4. RESULTS ON FISSION PRODUCT BEHAVIORS

Fig.4 shows release and deposition fractions of 12 RN classes to each compartment of plant. Fig.5 shows the release fractions from fuel, and deposition fractions to RPV/RCS and containment, and release to the environment of 12 classes. Fig.6 shows fission product distribution in each compartment at the simulation ending time, i.e. SBO at 72 h and SGTR at 20 h, respectively. Fission product distribution has a saturation behavior in each compartment at these simulation ending times. Table 6 shows the comparison of study results for OPR-1000 and estimation results of Fukushima Daiichi accident.

Fission product release behaviors are shown in Fig.4-1, 4-2, and 6-1 in SBO case, while fission product release behaviors in SGTR case are shown in Fig.4-2, 5-2, and 6-2. The fission product release to environment starts at 43 h in SBO case, while it starts at 4 h in SGTR case. In the SGTR case, RPV lower head failure occurs at 9 h. Even though release timing is different between SBO case and SGTR case, release patterns among RN classes are very similar. To estimate the impact of the fission product release to the environment, the initial inventory estimated by ORIGEN-ARP code for OPR-1000 (Ryu et al., 2016) should be multiplied to the environmental release fraction estimated in this study.

As shown in Fig.6-1 and 6-2, over 90% of inventory is released from fuel for class 1 (NG), 2(Cs), 4 (I), 5 (Te), 7 (Tc), 11 (Sb), and 12 (Ag) in both cases. However, less than 40% of inventory are released from fuel for less volatile or non-volatile classes 3 (Sr), 6 (Ru), 8 (Ce), 9 (Cm), and 10 (U). For class 6 (Ru), only 20% is released from fuel for SBO case while 62% is released from fuel, respectively. For class 7 (Mo, Tc, Fe), over 90% is released from fuel for SBO case while only 30% is released from fuel, respectively.

Over 90 % of noble gas (class 1) is released to environment in both cases. 76% and 67% of class 2 (Cs) are retained in RPV in SBO and SGTR cases, respectively. 7% of class 4 (I) is retained in both cases.

In SBO case, most of the classes (3 through 12), except classes 1 and 2, are retained in containment. In SGTR case, classes 3, 7, 8, 9, and 10 are retained in containment. These classes are released from fuel debris in the reactor cavity by MCCI after RPV lower head failure occurs at 9 h.

The release rate of fission product in Fukushima Daiichi Units 1, 2, and 3 cores are well summarized in chapter 4 of the book edited by Povinec et al. (2013). Activity release rates of I-131 and Cs-137 are summarized well in p.116 of UNSCEAR 2013 report, too. Total initial inventories of about 6000 PBq and 700 PBq of I-131 and Cs-137 are existed in the 3 Units cores. Among these, 120 PBq (2% of I-131 initial inventory) and 14 PBq (2% of Cs-137 initial inventory) are released to the atmosphere. The amount direct discharge of Cs-137 to the sea is 3.7 PBq (0.5% of initial Cs-137 inventory). About 30% and 20% of I-131 and Cs-137 are retained in the reactor building and turbine building. Each Unit has different release rate depending on the accident progression. It is reported that Unit 2 has highest release rate among 3 Units. Unit 1 has smallest

initial inventory due to lowest reactor power. Unit 2 and Unit 3 have the same initial inventories. Amount of initial inventories of fission products are calculated by ORIGEN-2 code (Nishihara et al. JAEA-Date/Code 2012-018).

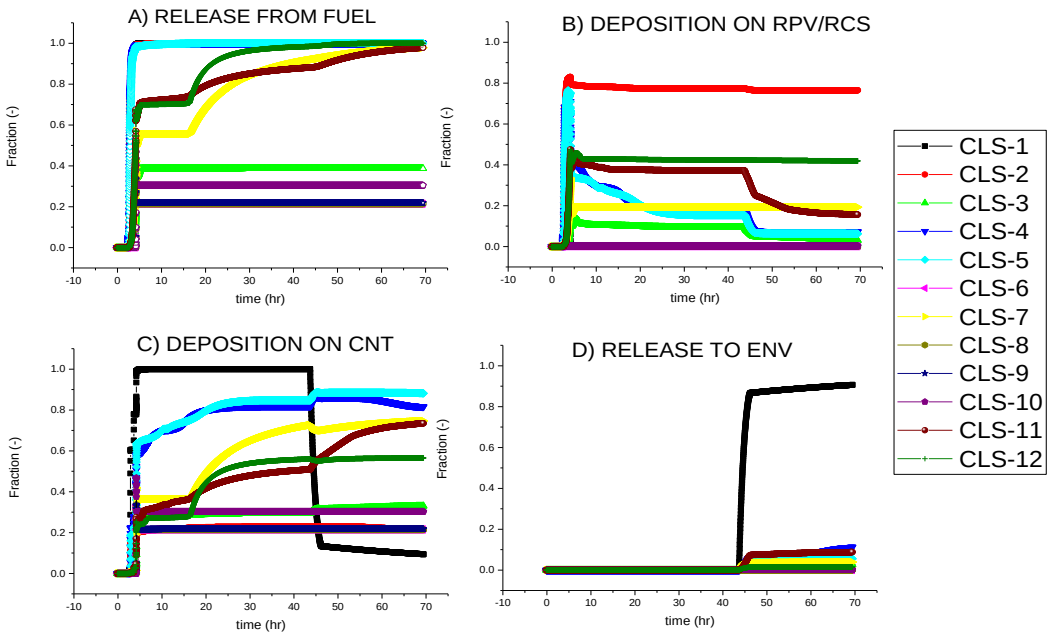

Fig.4-1 Release and deposition fractions of 12 RN classes in SBO case

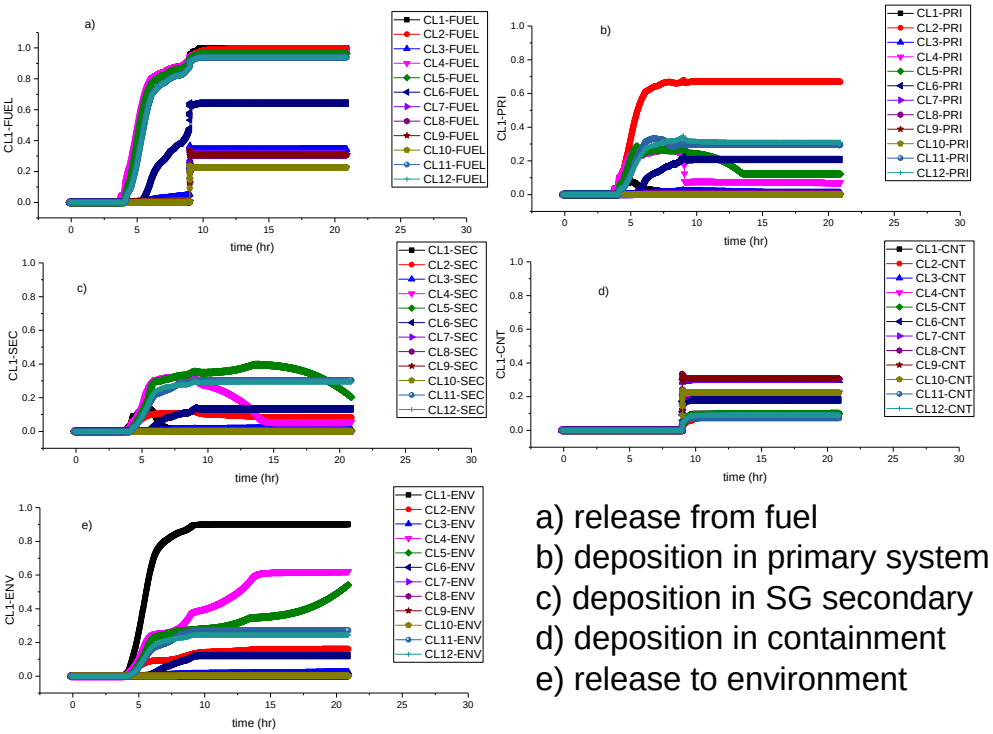

Fig.4-2 Release and deposition fractions of 12 RN classes in SGTR case

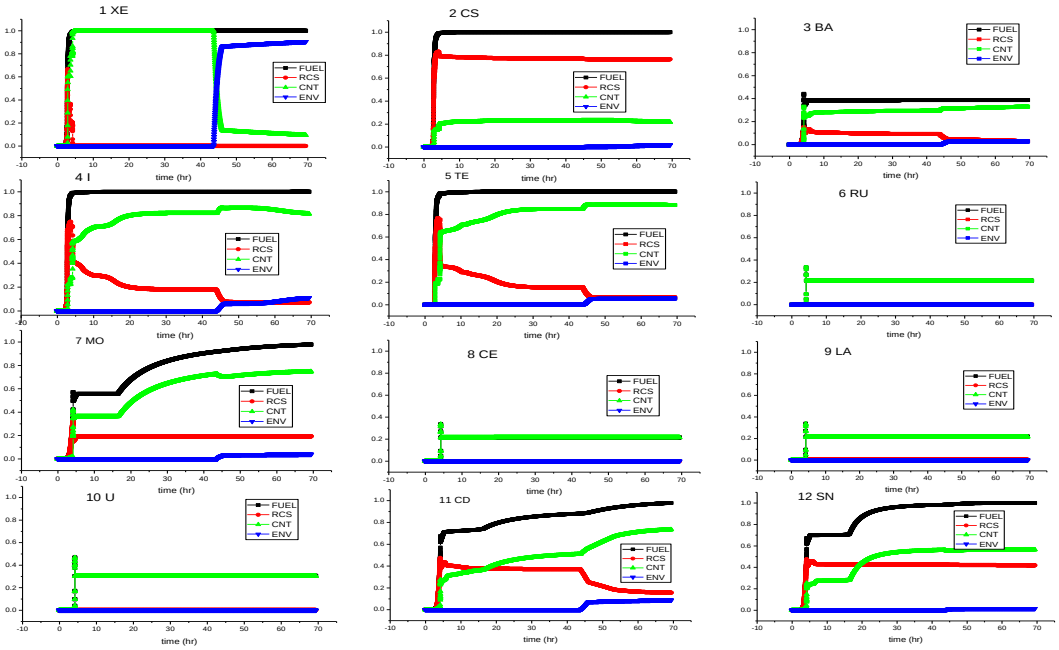

Fig.5-1 Release fractions of 12 RN classes in SBO case

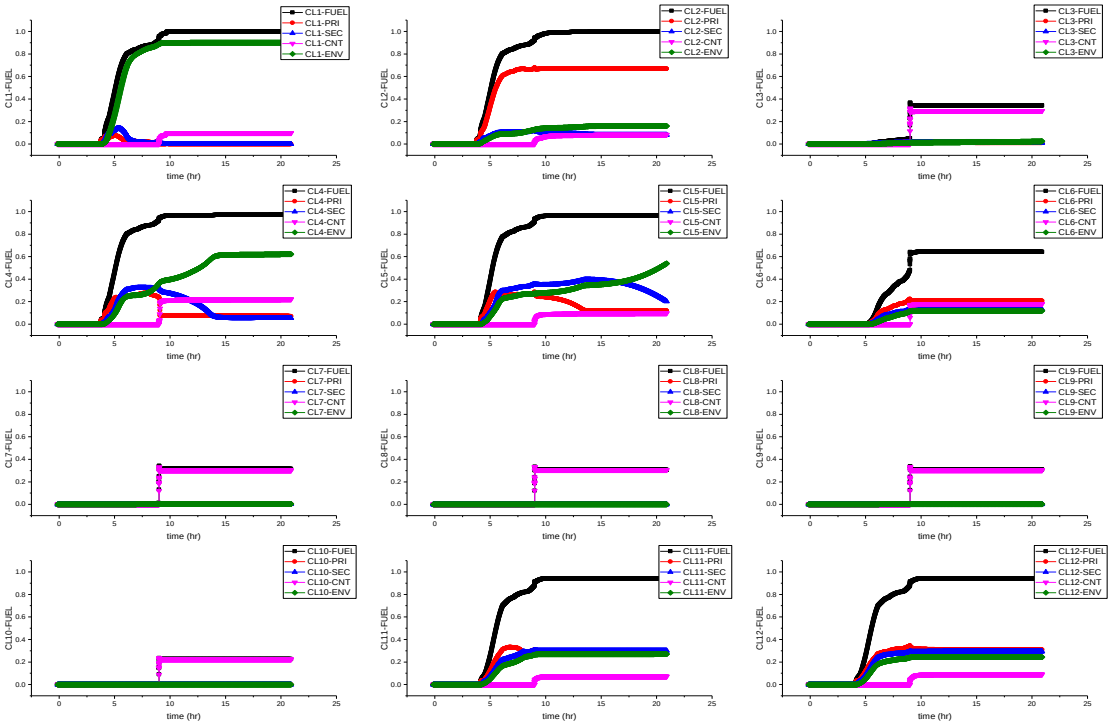

Fig.5-2 Release fractions of 12 RN classes in SGTR case

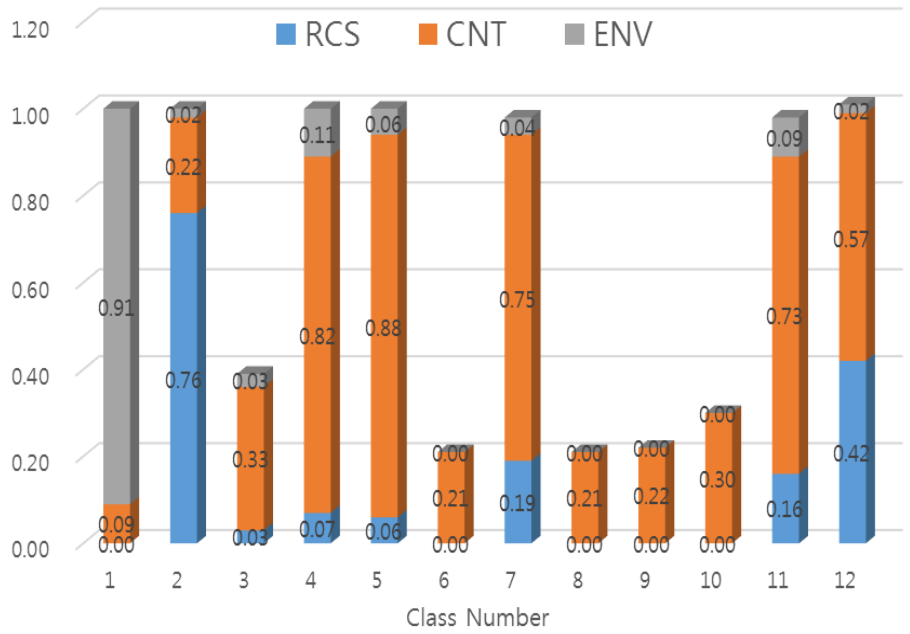

Fig.6-1 Compartmental Deposition Fractions in SBO Case at 70 h

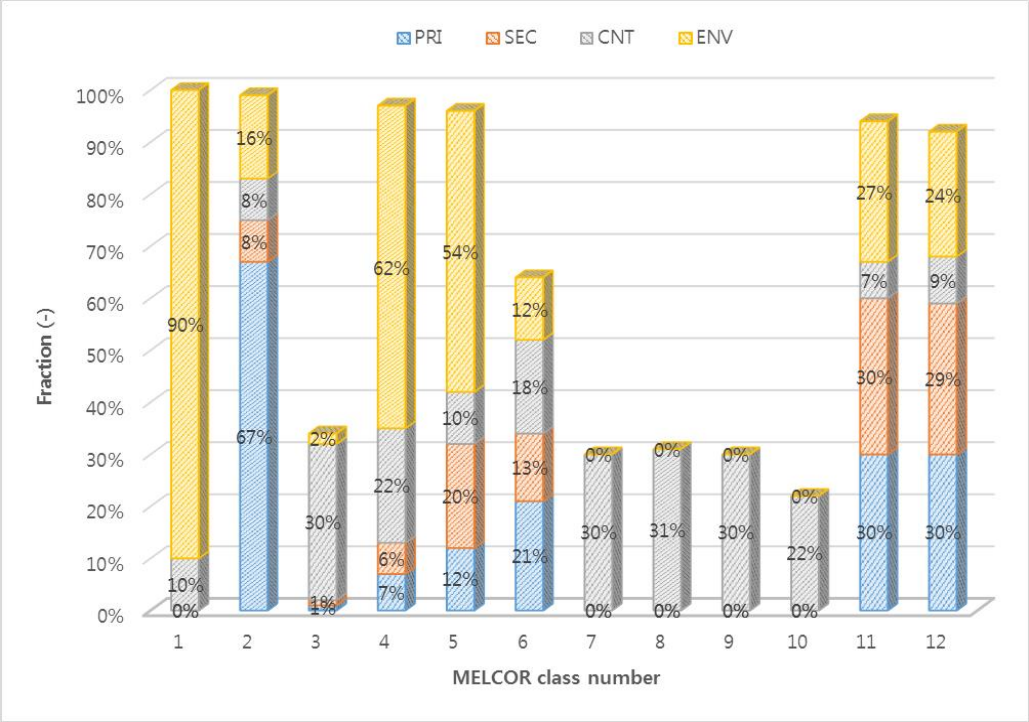

Fig.6-2 Compartmental Deposition Fractions in SGTR Case at 20 h

Table 6. Comparison of release fractions to environment of radionuclides between this study and Fukushima accident

MELCOR Class number	Typical Radio Nuclide	OPR-1000 SBO in this study	OPR-1000 SGTR in this study	UNSCEAR 2013 (Fukushima Daiichi)	Fukushima Daiichi, Released to Atmosphere[a]	Fukushima Daiichi, Released to Stagnant Water[b]	Fukushima Daiichi, Direct Discharge to Pacific Ocean[c]
1	Xe-133 Kr-85	91%	90%		100% assumed	0% assumed	
2	Cs-137	2%	16%	1 - 3 %	2.2%	20%	0.50%
	Cs-134				2.4%	20%	0.40%
3	Sr-89	3%	2%		0.033%	1.2%	
	Sr-90				0.027%	1.6%	1E-5%
4	I-131	11%	62%	2 - 8%	2.6%	32%	
	I-133				8.0%	-	
5	Te-129m	6%	54%		1.8%	-	
	Te-132				1.0%	-	
6	Ru-106	0%	12%				
7	Tc-99m	4%	0%				
8	Ce-144	0%	0%		0.00019%		
	Pu-238				0.00013%		
	Pu-239				0.00012%		
	Pu-240				0.00010%		
9	Cm-242	0%	0%		0.00004%		
11	Sb-125	9%	27%				
12	Ag-110m	2%	24%				

a) Povinec et al, 2013, p.118
b) Povinec et al, 2013, p.120, Hidaka 2014, Nishihara 2012
c) Tsumune et al. (2012)

II. CONCLUSIONS

MELCOR simulation for the OPR-1000 plant shows the following insights on the fission product behaviors on the severe accidents. In case of SBO sequence, 2% of cesium inventory and 11% of iodine inventory are released to the environment, while about 80% of iodine and about 21% of cesium of core inventories release to the environment in this study in SGTR case. 76% and 67% of cesium inventory are remained in RPV/RCS in SBO and SGTR cases, respectively. 82% and 22% of iodine inventory are remained in containment in SBO and SGTR cases, respectively.

About 2.2% of Cs-137 inventory and 2.6% of I-131 inventory are estimated to be released to the atmosphere in the Fukushima accident. 20% of Cs-137 inventory and 32% of I-131 inventory are estimated to be released to the stagnant water in reactor building or turbine building.

In the SBO case of OPR-1000 and Fukushima accident, they have the same order of magnitude, 2% of cesium inventory (2% in OPR-1000, 2.2% and 2.4 % of Cs-137 and Cs-134 in Fukushima), is released to the atmosphere. For iodine inventory, 11% in OPR-1000, and 2.6% and 8% of I-131 and I-133, respectively in Fukushima are estimated to the atmosphere. Roughly speaking, it can be said that they have similar results in iodine release to atmosphere, too.

Even though release timing information is different between SBO and SGTR cases in OPR-1000 severe accident scenarios, the release patterns among 12 RN classes have much similarities between the two accident scenarios.

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REFERENCES

1. Tokyo Electric Power Co., Inc. (TEPCO), Fukushima Nuclear Accident Analysis Report, June 20, (2012)
2. OECD/NEA, Benchmark Study of the Accident at the Fukushima Daiichi Nuclear Power Plant (BSAF) Project, Phase I, Final Report, March (2015)
3. T. Nishimura, H. Hoshi, A. Hotta, Current Research and Development Activities on Fission Products and Hydrogen Risk after the Accident at Fukushima Daiichi Nuclear Power Station, Nuclear Engineering Technology, 47 (2015) 1-10
4. Sandia Report, SAND2012-6173, Fukushima Daiichi Accident Study (Status as of April 2012), Sandia National Laboratories, USA, printed August (2012)
5. United Nations Scientific Committee on the Effects of Atomic Radiation, UNSCEAR 2013 Report, Volume I, Report to the General Assembly, Scientific Annex A: Levels and effects of radiation exposure due to the nuclear accident after the 2011 great east-Japan earthquake and tsunami, (2013)
6. World Health Organization (WHO), Health risk assessment from the nuclear accident after the 2011 great east Japan earthquake and tsunami based on a preliminary dose estimation (2013)
7. World Health Organization (WHO), Preliminary dose estimation from the nuclear accident after the 2011 great east Japan earthquake and tsunami (2012)
8. KEPRI, "Probabilistic safety assessment for Ulchin Units 5&6 (phase Ⅱ): containment performance analysis (final report)," Korean Electric Power Corporation, (in Korean language) (2002).
9. KHNP, OPR1000 Plant Description, (http://www.opr1000.co.kr/), Korea Hydro and Nuclear Power, (2014).
10. US NRC, "MELCOR Computer Code Manuals," NUREG/CR-6119. Sandia National Laboratories, (2005).
11. US NRC, "State-of-the-Art Reactor Consequence Analyses (SOARCA) Main Report," NUREG-1935, (2012).
12. US NRC, "State-of-the-Art Reactor Consequence Analyses (SOARCA) Project, Volume 2: Surry Integrated Analysis," Sandia Nation Laboratories, US NRC, NUREG/CR-7110, Vol. 2. (2012).
13. US NRC, "Reassessment of the Technical Bases for Estimating Source Terms", NUREG-0956. (1986).
14. US NRC, "Radionuclide Release Calculations for Selected Severe Accident Scenarios", NUREG/CR-4624, BMI-2139, Vol.5. PWR, Large Dry Containment Design, (1986).
15. US NRC, "Reactor Safety Study, An Assessment of Accident Risks in the U.S. Commercial Nuclear Power Plants", WASH-1400, NUREC-75/014, (1975).
16. Min Lee, Y-Chen Ko, "Quantification of severe accident source terms of a Westinghouse 3-loop plant", Nuclear Engineering and Design (NED), Vol. 238, 1080-1092, (2008).
17. J.J. DiNunno, et al., "Calculation of Distance Factors for Power and Test Reactor Sites," TID-14844, U.S. Atomic Energy Commission (now USNRC), (1962).
18. L. Soffer et al., "Accident Source Terms for Light Water Nuclear Power Plants," NUREG-1465, U.S. Nuclear Regulatory Commission, (1995).
19. Miwa et al, Research Program for the evaluation of fission product and actinide release behavior, focusing on their chemical forms, Energy Procedia 71 (2015) 168-181
20. Wright, A.L., Primary System Fission Product Release and Transport, A state-of-the-art report to the committee on the safety of nuclear installations (CSNI), NUREG/CR-6193, NEA(CSNI/R(94)2, ORNL/TM-12681, (1994)
21. Povinec P.P, Hirose K., Aoyama M. edited, Fukushima Accident: Radioactivity Impact on the Environment, (2016), Elsevier
22. Nishihara K., Iwamoto H., Suyama K., JAEA-Date/Code 2012-018, Estimation of Fuel Compositions in Fukushima Daiichi Nuclear Power Plant
23. Nishihara K., et al., Radionuclide Release to Stagnant Water in Fukushima-1 Nuclear Power Plant, Atomic Energy Society of Japan, 2012, in Japanese
24. Hidaka A., Ishikawa J., Quantities of I-131 and Cs-137 in accumulated water in the basements of reactor buildings in process of core cooling at Fukushima Daiichi nuclear power plants accident and its influence on late phase source terms, J. Nuc. Sci. Tech., 2014 Vol.51 No.4 413-424
25. Tsumune, D., Tsubono, T., Aoyama, M., Hirose, K., 2012. Distribution of oceanic 137Cs from the Fukushima Daiichi Nuclear Power Plant simulated numerically by a regional ocean model. J. Environ. Radioact. 111, 100-108.
26. C.V. McIsaac, R. Kohli, R.S. Denning et al., Fission Product Behavior Distribution in the TMI-2 Reactor Building, EGG-M-88202, Idaho National Engineering Laboratory (1988)

27. D.W. Akers, R.K. McCardell, M.L. Russell et al., TMI core materials and fission product inventory, Nucl. Eng. Des., 118, 451-461 (1990)
28. UNSCEAR 2008 Human Health Risk Assessment for Chernobyl Accident
29. Kim T.W. et al., KAERI/AR-1099/2015 Review of International and Domestic Source Term Research Histories for the Establishment of Radiological Emergency Preparedness Framework of Nuclear Power Plant
30. Kim T.W. et al., KAERI/RR-6266/2015 Development and Application of Source Term Analysis Framework on OPR-1000 NPP for Radiological Emergency Preparedness
31. Ryu Y.H, Ha K.S, Kim T.W, Kim D.H, KAERI/TR-6425/2016, Calculation of the Fission Products in the OPR1000 using ORIGEN-ARP Code
32. Buesseler K., Di M., Aoyama M., et al., (2016), Fukushima Daiichi-Derived Radionuclides in the Ocean: Transport, Fate, and Impacts, Annu. Rev. Mar. Sci.
33. Steinhauser G., Fukushima's Forgotten Radionuclides: A Review of the Understudied Radioactive Emissions, Envir. Sci. Technol., 2014, 48 4649-4663
34. Lin W.H., Chen L.Q, Yu W., Ma H., Zeng Z., Zeng S., Radioactive source terms for the Fukushima nuclear accident, Science China Earth Sciences, Jan. 2016, Vol.59, No.1, 214-222
35. Shozugawa K., Norio N., Matsuo M., Deposition of fission and activation products after the Fukushima Daiichi nuclear power plant accident, Environmental Pollution, 163 (2012) 243-247
36. Evrard O., Laceby J.P., Lepage H., Onda Y., Cerdan O., Ayrault S., Radiocesium transfer from hillslopes to the Pacific Ocean after the Fukushima Nuclear Power plant accident: A Review, J. Env. Radio., 148 (2015) 92-110
37. Aoyama M., Kajino M., Tanaka T.Y., Sekiyama T.T., Tsumune D., Tsubono T., Hamajima Y., Inomata Y., Gamo T., 134 Cs and 137 Cs in the North Pacific Ocean derived from the March 2011 TEPCO Fukushima Daiichi Nuclear Power Plant accident, Japa. Part two: estimation of 134Cs and 137Cs inventories in the North Pacific Ocean, J. Oceanogr (2016) 72:67-76
38. Katata G., Chino M., Kobayashi T., Terada H., Ota M., Nagai H., Kajino M., Draxler R., Hort M.C., Malo A., Torii T., Sanada Y., Detailed source term estimation of the atmospheric release for the Fukushima Daiichi Nuclear Power Station accident by coupling simulations of an atmospheric dispersion model with an improved deposition scheme and oceanic dispersion model, Atmos. Chem. Phys., 15 1029-1070, (2015)