In-Plant Fission Product Behavior in SGTR Accident with Long-Term SBO

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1. Introduction

Since the Three Mile Island (TMI) (1979), Chernobyl (1986), Fukushima Daiichi (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: 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 Steam Generator Tube Rupture (SGTR) for Surry plant 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 accident, the assessment of radiological source term effects on the environment has been a revitalized key concern of nuclear safety.

Containment bypass sequences such as SGTR and Interfacing Systems LOCA (ISLOCA), even though their occurrence frequency is smaller than other sequences such as LOCA and loss of feedwater transients, can result in serious hazard to environment. 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 Fukushima accident, off-site and on-site AC powers were lost by tsunami attack about 45 minutes after earthquake. DC 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 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. The off-site AC power was recovered in 9 days after the accident in the NPS. Therefore safety injection by fire pump truck with fresh water or seawater is only available in the Fukushima accident. However, safety injection by fire pump truck is not always effective due to the high pressure of RPV inside or leakages of alternative water injection flow paths.

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.

Iodine pool-scrubbing can occur in the secondary side of the faulted steam generator. However, iodine pool-scrubbing in the secondary side of the faulted steam generator is assumed not to be working, due to the assumption of the loss of DC battery for turbine inlet flow control valve. Iodine pool-scrubbing is one of the long-term research issues in safety assessment of nuclear power plant severe accident. PHEBUS FPT series and THAI experiment projects are typical projects on the resolving source term issues in severe accident of nuclear power plants. However, iodine retention by pool scrubbing is still a debating issue.

In such containment bypass sequences, fission products can be released out to environment directly from RCS without retention or deposition in containment structures. SGTR is one of the hazardous accident scenarios in the typical PSA, because SGTR induces a large release amount of source term to environment directly. For SGTR, there are considerable variations of accident progression due to operation of safety features and operation strategies within the consequence of core damage. The characterizing of accident progression of SGTR for the estimation of radiological source term is a key aspect.

This study focuses on SGTR accident progression affected on the source term behaviors. The characteristics of SGTR were estimated by varying the key safety features and plant operation strategies. A key operation strategy was the isolation of the broken reactor coolant system loop from the intact loop. Typical core degradation in SGTR scenarios occurs with multiple failures of the isolation, secondary cooling and safety injection strategy of RCS. Particularly, the isolation failure provides release path of radiological source term to the environment. With the given scenarios, in-plant fission product behaviors are estimated by using MELCOR code version 1.8.6.

It is necessary to study a more detailed SGTR considering its importance in the consequential effects, but there are a few of knowledge bases of radiological source term behaviors during SGTR.

2. MELCOR MODELING FOR SGTR

2.1 MELCOR NODALIZATION

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 reference plant is shown in Fig. 1.

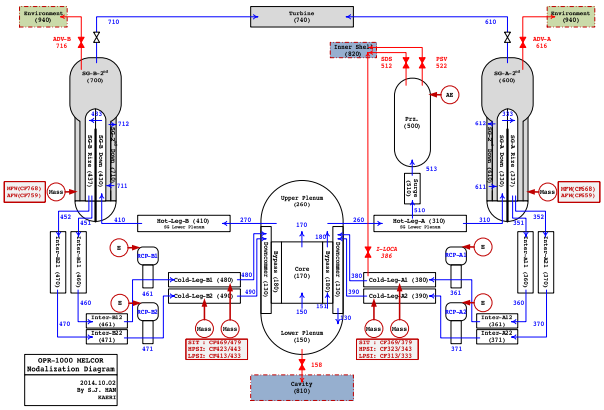

Fig. 1 MELCOR CVH/FL nodalization diagram for OPR-1000, typical Korean PWR

The elevations of control volumes are set from reference level of hot leg centreline (0.0 m). The total coolant inventory of RCS except pressurizer volume is about 288 m3. Lower Plenum, Core, and Bypass control volumes are linked with COR package. COR cells consist of 13 levels and 7 radial rings. Core materials which can be molten during severe accident scenarios are 85.6 tons of fuel, 23.9 tons of zircaloy cladding, and 11.7 tons of core supporting structural material of stainless steel.

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.

Table 1. Initial Mass of Core Materials

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

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.

Only four passive safety injection tanks (SIT) with total 200 tons of water reserved. The water is automatically injected into the RCS when RCS pressure drops below 4 MPa.

Table 2. 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 3 shows key events of given scenario. Core uncovery (TAF) occurs at 3.25 hr. 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 occurs during from 4 to 10 hr. Finally failures of lower head penetrations occur from 10.3 hr to 11.5 hr.

Table 3. Accident Progression of Given Scenario

event	time(s)	time(hr)
SG-A SGTR occurs	0	
SG-A ADV open	10	
Reactor trip, MFW trip, MSIV close	10	
Core uncovery (TAF) occurs	11708	3.25
SG-B Dryout	12118	3.37
SG-A Dryout	13526	3.76
Gap release	13597	3.78
Spacer grids starts to collapse	14718	4.09
Start of cladding melting	14770	4.10
Start of debris quench	34897	9.69
Core support plate (CSP) failure starts	36894	10.25
Core dryout (BAF)	37078	10.30
Ring 1 Penetration Failure	41119	11.42
Ring 2 Penetration Failure	41337	11.48
Ring 3 Penetration Failure	38654	10.74
Ring 4 Penetration Failure	37076	10.30
Ring 5 Penetration Failure	37073	10.30
Ring 6 Penetration Failure	37110	10.31
Ring 7 Penetration Failure	37275	10.35

3. Results and Discussion

3.1. Thermal Hydraulic Behaviors

3.1.1. Liquid flow through ADV

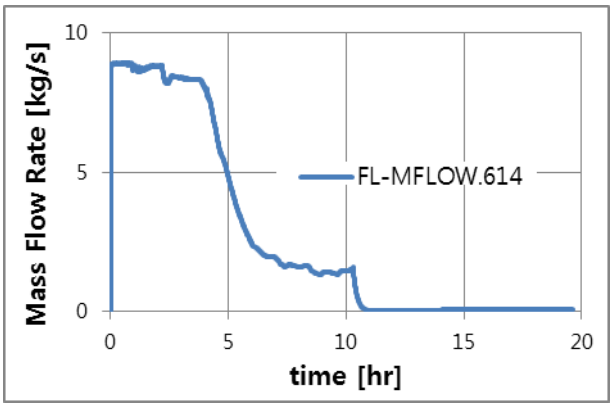

Fig.2 Mass flowrate through ADV

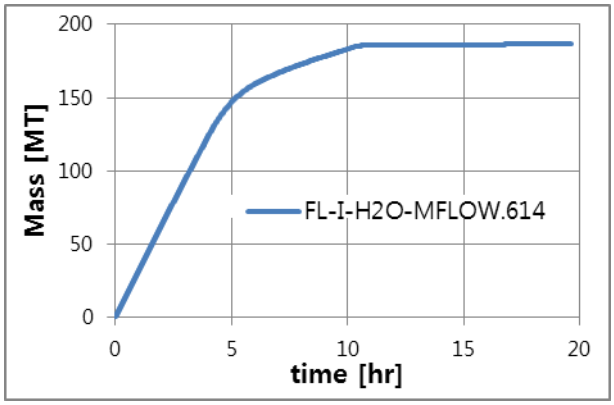

Fig.3 Integrated mass flowate through ADV

Fig.2 shows liquid mass flow rate through ADV to environment. About 9 kg/s of mass flow rate, the liquid flew out to environment up to 4 hr. About 1.5 kg/s of mass flow rate, the liquid flew out to environment from 6 hr to 10.3 hr.

Fig.3 shows integrated liquid flow mass to environment through ADV. Total 186 tons of initial RCS water inventory flew out to environment through ADV up to time 10.3 hr when vessel failure occurs.

3.1.2. RCS Behaviors

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

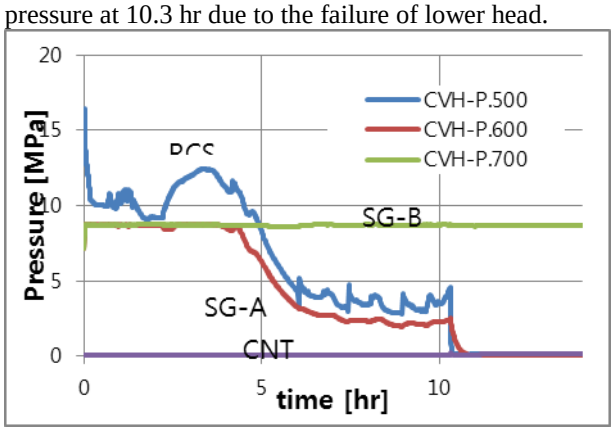

Fig.4 RCS and Secondary Pressure

Fig.5 shows RPV water level transients. Core uncover (TAF) occurs at 3.25 hr 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 hr. Core materials slumping down to lower plenum from 6 hr to 10.3 hr.

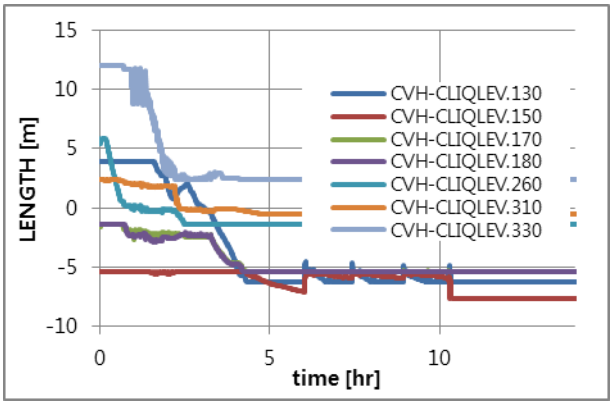

Fig.5 RCS Water level s at each control volume

Fig.6 shows liquid temperature of RPV control volumes. Liquid temperature in core channel (CV170) start to increase at 4 hr and reaches above 2500 C at about 7 hr. Fuel melting starts to occur at this time.

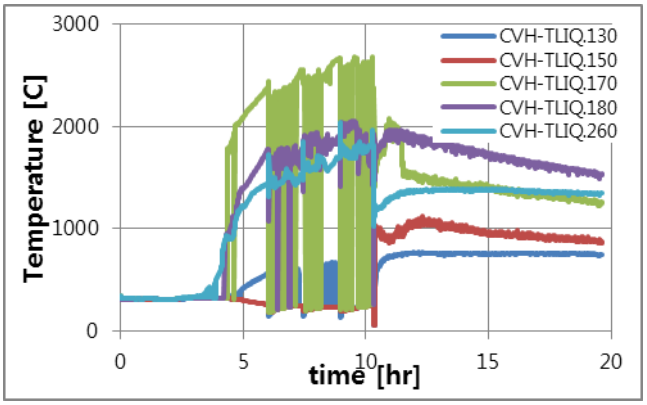

Fig.6 Liquid Temperature of RPV Control volumes

3.2. Source Term Behaviors in RCS and Containment

Fission products may be aerosolized as they are released from fuel early in a light water reactor (LWR) accident and later expelled from the reactor coolant system. Other events and processes that occur late in the accident, such as core-concrete interactions, pool boiling, direct containment heating, deflagrations, and resuspension may also generate aerosols. High structural temperatures may also result in aerosolization of nonradioactive materials. Most of the radioactive material that can escape from a nuclear power plant during a severe reactor accident will do so in the form of aerosols. Much of reactor accident analysis is the prediction of the behavior of these radioactive aerosols. Aerosols are very small solid particles or liquid droplets suspended in a gas phase.

The suspended solid or liquid particles typically have a range of sizes. Minimum and maximum default sizes of aerosol particles range in lognormal distribution from 0.1 μm to 50 μm in the MAEROS code. 20 size bins are used in the agglomeration process. Aerosol concentrations in reactor accident analyses are typically less than 100 g/m^3 and usually less than 1 g/m^3. In the ABCOVE AB6 test, for example, maximum suspended NaOx aerosol concentration measure was 33 g/m^3, reached 600s after the initiation of the sodium spray. A steady-state concentration of 28g/m3 was attained by the end of the NaOx source period. The suspended Nal concentration attained a maximum value of 0.27g/m3, and then slowly declined to 0.085g/m3 by the end of the Nal source.

In MELCOR code, RN (Radionulcide) 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 parameters obtained from the HS (Heat Structure) package are:

- Geometric orientation
- Surface area in the atmosphere
- Surface heat flux
- Mass transfer coefficient
- Water condensation mass flux

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 phoretic 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.

3.2.1. Fission Product Release from Fuel

In MELCOR code version 1.8.6, there are 16 aerosol classes treated, which is shown in Table 4.

Gap release from the fuel cladding gap :
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. When any axial position in a fuel bundle achieves a temperature of 900 C, CORSOR calculates a gap release of certain volatile fission products for all fuel rods in that radial zone. The amount of gap release is taken to be 5% of the initial amount present for cesium, 1.7% for iodine, 3% for noble fission gases such as Xe and Kr, 0.01% for Tellurium and antimony, and 0.0001% for barium and strontium. However, this emission is very small in comparison with the melt release.

Melt Release from Fuel Pellet :
Three options are currently available in MELCOR code for the release of radionuclides from the core fuel component: CORSOR, CORSOR-M, CORSOR-BOOTH model. The computation of the fractional release rate coefficients for fission products is based on empirical correlations derived from experiments (NUREG/CR-0722, NUREG-0772, NUREG/CR-1288, NUREG/CR-1386, NUREG/CR-1773, etc.). The same correlation is used to calculate the release rate for a given class using the individual temperature of that component. That is, the calculation of release of radionuclides from fuel, cladding, canisters, control rods, and particulate debris differs only in the temperature used. Separate correlations for these components are not employed since their form is not compatible with the MELCOR structure.

Table 4. 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

1) CORSOR model

$$\dot{f} = A\exp(BT) \qquad for\ T \geq T_i \qquad (1)$$

The original CORSOR model correlates the fractional release rate in exponential form. A and B are empirical coefficients based on experimental data, and T is the core cell component temperature in degrees Kelvin. Different values for A and B are specified for three separate temperature ranges. The lower temperature limit Ti (900 K) for each temperature range and the A and B values for that range are defined for each class in sensitivity coefficient array 7101. Fission products start migration from fuel pellet matrix to the gap when fuel temperature reaches to 900K.
In case of class 5 fission product (Te) release rate has different coefficients among three separate temperature ranges (T>900K, 900K<T<1400K, T>2200K) while the other classes except Te have different coefficients among three separate temperature ranges (T>900K, 900K<T<1600K, T>2000K). Different temperature ranges and different release coefficients (A and B in eq.(1)) are shown in Table 5 and 6, respectively.

Table 5. Temperature criterion in CORSOR model

	C7101(1,1,k)	C7101(2,1,k)	C7101(3,1,k)
All Classes except Class 5	900.E0	1400.E0	2200.E0
	C7101(1,1,5)	C7101(2,1,5)	C7101(3,1,5)
Class 5	900.E0	1600.E0	2000.E0

Table 6. Coefficients A and B according to class and temperature in CORSOR model

		C7101(1,j,k)	C7101(2,j,k)	C7101(3,j,k)
Class 1	A	7.02E-9	2.02E-7	1.74E-5
	B	0.00886	0.00667	0.00460
Class 2	A	7.53E-12	2.02E-7	1.74E-5
	B	0.0142	0.00667	0.00460
Class 3	A	7.50E-14	8.26E-9	1.38E-5
	B	0.0144	0.00631	0.00290
Class 4	A	7.02E-9	2.02E-7	1.74E-5
	B	0.00886	0.00667	0.00460
Class 5	A	1.62E-11	9.04E-8	6.02E-6
	B	0.0106	0.00552	0.00312
Class 6	A	1.36E-11	1.36E-11	1.40E-6
	B	0.00768	0.00768	0.00248
Class 7	A	5.01E-12	5.93E-8	3.70E-5
	B	0.0115	0.00523	0.00200
Class 8	A	6.64E-12	6.64E-12	1.48E-7
	B	0.00631	0.00631	0.00177
Class 9	A	5.00E-13	5.00E-13	5.00E-13
	B	0.00768	0.00768	0.00768
Class 10	A	5.00E-13	5.00E-13	5.00E-13
	B	0.00768	0.00768	0.00768
Class 11	A	1.90E-12	5.88E-9	2.56E-6
	B	0.0128	0.00708	0.00426
Class 12	A	1.90E-12	5.88E-9	2.56E-6
	B	0.0128	0.00708	0.00426
Class 13 to 20	A	0.	0.	0.
	B	0.	0.	0.

2) CORSOR-M model

$$\dot{f} = k_0\exp(-Q/RT) \qquad (2)$$

The CORSOR-M model correlates the same release data used for the CORSOR model using an Arrhenius form. The values of k and Q for each class are implemented in sensitivity coefficient array 7102 in MELCOR code and they are summarized again in Table 7.

Table 7. CORSOR-M coefficients

Class	k_o (C7102(1,i))(1./min)	Q (C7102(2,i))(kcal/mole)
1	2.00E5	63.8
2	2.00E5	63.8
3	2.95E5	100.2
4	2.00E5	63.8
5	2.00E5	63.8
6	1.62E6	152.8
7*	23.15*	44.1*
8 9**	2.67E8 1.46E7 **	188.2 143.1**
10	1.46E7	143.1
11**	5.95E3**	70.8**
12	5.95E3	70.8
13-30	0.	0.

3) CORSOR-Booth model

The CORSOR-Booth model considers mass transport limitations to radionuclide releases and uses the Booth model for diffusion with empirical diffusion coefficients for cesium releases. Release fractions for other classes are calculated relative to that for cesium.

$$\dot{f}_k\left(\frac{fraction}{s}\right) = \frac{\dot{m}_{tot,k}}{\rho V}\left[F - \frac{P_{k,bulk}}{P_{k,eq}}\right] \quad (3)$$

$$\dot{m}_{tot,k} = \frac{1}{DIFF_k^{-1} + \dot{m}^{-1}} \quad (4)$$

$$DIFF_k = DIFF_{Cs} S_k \quad (5)$$

Nominal values for S_k are given in sensitivity coefficient array 7103 in MELCOR. For certain conditions of cladding oxidation and temperature, the scaling factors must be modified for some classes. When the oxide mass fraction exceeds a critical value F_{k1} and the temperature exceeds a critical value T_{k1}, the class scaling factor is given by equation. When the oxide mass fraction is below a minimum value F_{k2}, the class scaling factor is given by S_{k2}. The release rate for species other than cesium is given by multiplying the cesium release rate by an appropriate scaling factor for each RN class in MELCOR.

$$S_k = S_{k1}\exp(C_k T) \quad or \quad S_{k2} \quad (6)$$

The diffusion release rate for species other than cesium is given by multiplying the cesium release rate by an appropriate scaling factor S_k for each RN class k in MELCOR.

Table 8. CORSOR-Booth class scaling factors in MELCOR

Class	C7103(k)	Class	C7103(k)
1	1.0	8	3.34E-5
2	1.0	9	1.0E-4
3	3.33E-3	10	1.0E-4
4	1.0	11	5.0E-2
5	1.0	12	5.0E-2
6	1.0E-4	13-30	0.0
7	1.0E-3		

3.2.2. Core Materials Relocation Behavior

As the major concern of the current study, core materials and source term behaviors are shown in Fig. 7 through Fig. 11. Firstly, major escape mechanisms of radiological materials from the reactor core are related to severe accident phenomena. Particularly, core degradation mechanisms govern a primary release of source term and the transport of radiological materials through several compartment of the plant is related to the final environmental release.

Fig.7 shows maximum temperatures of core materials. Core heatup starts at 4 hr and ends at 10.3 hr when vessel failure occurs. Fuel cladding failure, support structure failure and core material relocation occurs during this time.

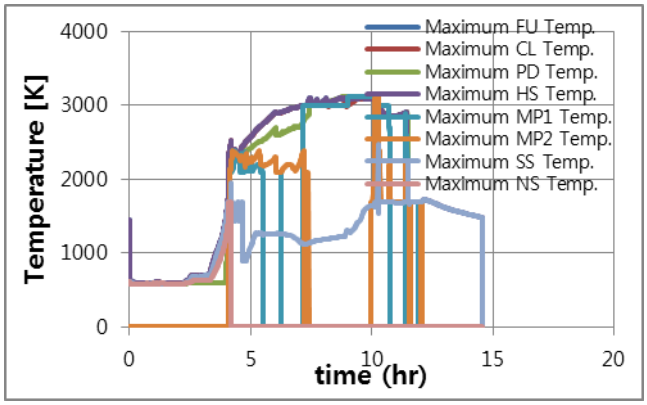

Fig.7 Maximum temperatures of core materials

Fig.8 shows key features of the core degradation. Core degradation materials consist of UO2, Zr, Stainless Steel, etc and relevant oxidations increases according to severe accident progression. Zr is changed to ZrO2 at 4 hr. Core materials ejected to cavity from 10.3 hr to 11.4 hr. Core region is modeled as 7 concentric radial rings.

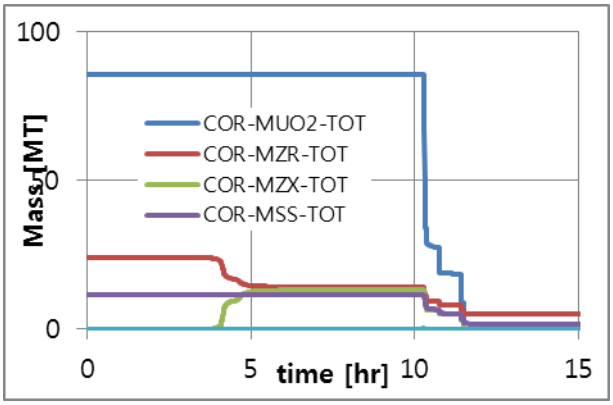

Fig.8 Mass changes of core materials

Fig.9 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) represent the transport amount of radiological materials leaving the core region to other compartments. After the vessel breach at 10.3 hr, 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 hr to 11.5 hr, which contributes long term behaviors of source term but add limited amounts comparing with the core degradation period in the current SGTR case.

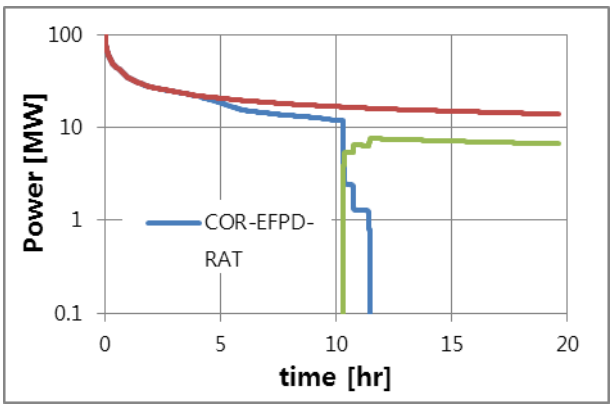

Fig.9 Decay heat distributions

3.2.3. Volatile Fission Products Behaviors in RCS

During core degradation, a large portion of radiological and non-radiological materials is generated as a vapor or aerosols. These aerosols move around the RCS and disperse to the environment through the faulted steam generator. Aerosols can be generated in the reactor cavity by MCCI process.

The natural attenuation of radioactive material available for release from nuclear power plants during accidents occurs because aerosol particles will deposit on surfaces in the reactor pressure vessel and RCS. Aerosols deposit on surfaces because they cross stream lines of flow over the surfaces or because they extend far enough to intercept the surface even when the particle center of mass is following a streamline. The rates of aerosol deposition on surfaces are often characterized in terms of 'deposition velocities' which are coefficients that relate the particle flux to the particle concentration in the gas phase. Processes that can lead to particle deposition include:

- gravitational settling
- diffusion to surfaces
- turbulent deposition
- inertial deposition

Inertial deposition process is not handled in MELCOR code currently.

Fig.10 and Fig.11 show the release fractions of Cesium and Iodine to various compartments of plant or to the environment through ADV opening.

Meanwhile, radiological materials transport on the SGTR release pathway is almost blocked by the vessel breach, because material source of release pathway of SGTR, i.e., core materials in vessel, are removed by the vessel breach. Fig.10 and Fig.11 show the source term environmental release fraction of this simulation case. As shown in Fig.10 and Fig.11, major portions of release fractions of source term are concentrated on from core degradation to vessel breach. After vessel breach between 10.3 to 11.5 hr, about 5.3% of Cesium and 6.8% Iodine are generated by MCCI at cavity.

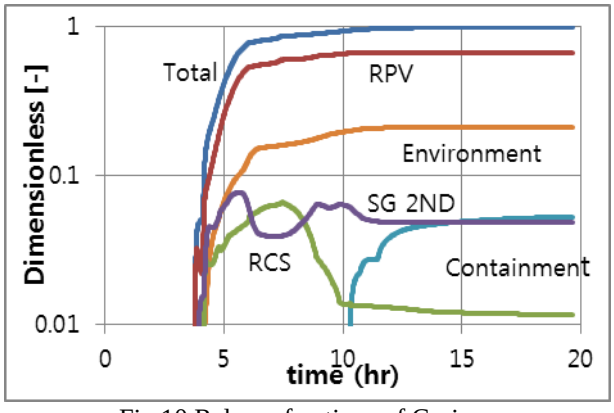

Fig.10 Release fractions of Cesium

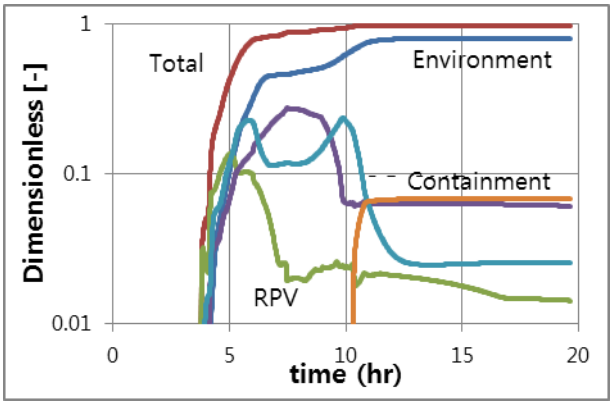

Fig.11 Release fractions of Iodine

Table 10 shows Cs and I Released and Deposited Fractions (%) at 20 hr.

In Cesium case, 99.5% of initial core inventory is released from fuel. Among these, 21% is released to environment through SG ADV, 5% is deposited in the SG secondary side, 67% is deposited in the RPV, 1% is deposited in the RCS, 5% is deposited in the containment.

In Iodine case, 96% of initial core inventory is released from fuel. Among these, 80% is released to environment through SG ADV, 2.5% is deposited in the SG secondary side, 1.4% is deposited in the RPV, 6% is deposited in the RCS, 6.8% is deposited in the containment.

The reason of 67% of Cesium retained in RPV while only 1.4% of Iodine retained in RPV is that Cesium is behaves as CsOH while Iodine behaves as CsI. CsI is much volatile than CsOH.

Table 10. Cs and I Released and Deposited Fractions (%) at 20 hr

Deposited place	Cesium (CsOH)	Iodine (CsI)
Deposited in RPV	67.1	1.4
Deposited in RCS	1.2	6.0
Deposited in SG secondary	4.8	2.5
Deposited in Containment	5.3	6.8
Released to Environment	21.1	79.5
Total release from Fuel	99.5	96.2

4. CONCLUDING REMARK

SGTR scenario for Surry plant treated in SOARCA project presented much reduced release amounts of source term than previous accident source term study results (TID-14844, NUREG-1465, NUREG-0956, etc.).

The release of major radioactive materials (Iodine and Cesium) were estimated as about 80% of Iodine and about 21% of Cesium of total core inventories release to environment in this study.

The reason of Iodine release fraction to environment (80%) is much greater than Cesium release fraction to environment (21%) is that 67% of Cesium retained in RPV while only 1.4% of Iodine retained in RPV. Cesium is behaves as CsOH while Iodine behaves as CsI after release from fuel. CsI is much volatile than CsOH.

Our results are almost equivalent to WASH-1400 results. In the PWR1 release category of WASH-1400, that is SGTR scenario, 70% of I and 40% of Cs are released to environment.

Currently, there is a large gap of the current application skill of source term in the Korean PSA comparing with SOARCA. The authors believe that key differences are due to (1) we used worst case scenario in this study while SOARCA used more optimal scenario, (2) the absence of modeling of cesium-molybdate (Cs_2MoO_4), (3) the shortage of deposition mechanism model in the SGTR flow path and (4) no application of pool scrubbing model appropriately, etc.

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