# Estimation of Source Term Behaviors in SBO Sequence in a Typical 1000MWth PWR and Comparison with Other Source Term Results

Tae Woon Kim [a*], Seok Jung Han [a], Kwang Il Ahn [a], Douglas Fynan [a], Yong Hoon Jung [a]

[a]Korea Atomic Energy Research Institute
*989-111 Daedeok-daero, Yuseong-gu, Daejeon, Korea, 305-353*
[*]*Corresponding author: twkim2@kaeri.re.kr*

Abstract – 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. In the Fukushima Daiichi accident, the long-term SBO (station blackout) accident occurs. Using the worst case assumptions like in Fukushima accident on the accident sequences and on the availability of safety systems, the thermal hydraulic behaviors, core relocation and environmental source terms behaviors are estimated for long-term SBO accident for OPR-1000 reactor. MELCOR code version 1.8.6 is used in this analysis. Source term results estimated in this study is compared with other previous studies and estimated results in Fukushima accidents in UNSCEAR-2013 report. This study estimated that 11 % of iodine can be released to environment and 2% of cesium can be released to environment. UNSCEAR-2013 report estimated that 2 ~ 8 % of iodine have been released to environment and 1 ~ 3 % of cesium have been released to the environment. They have similar results in the aspect of release fractions of iodine and cesium to environment.

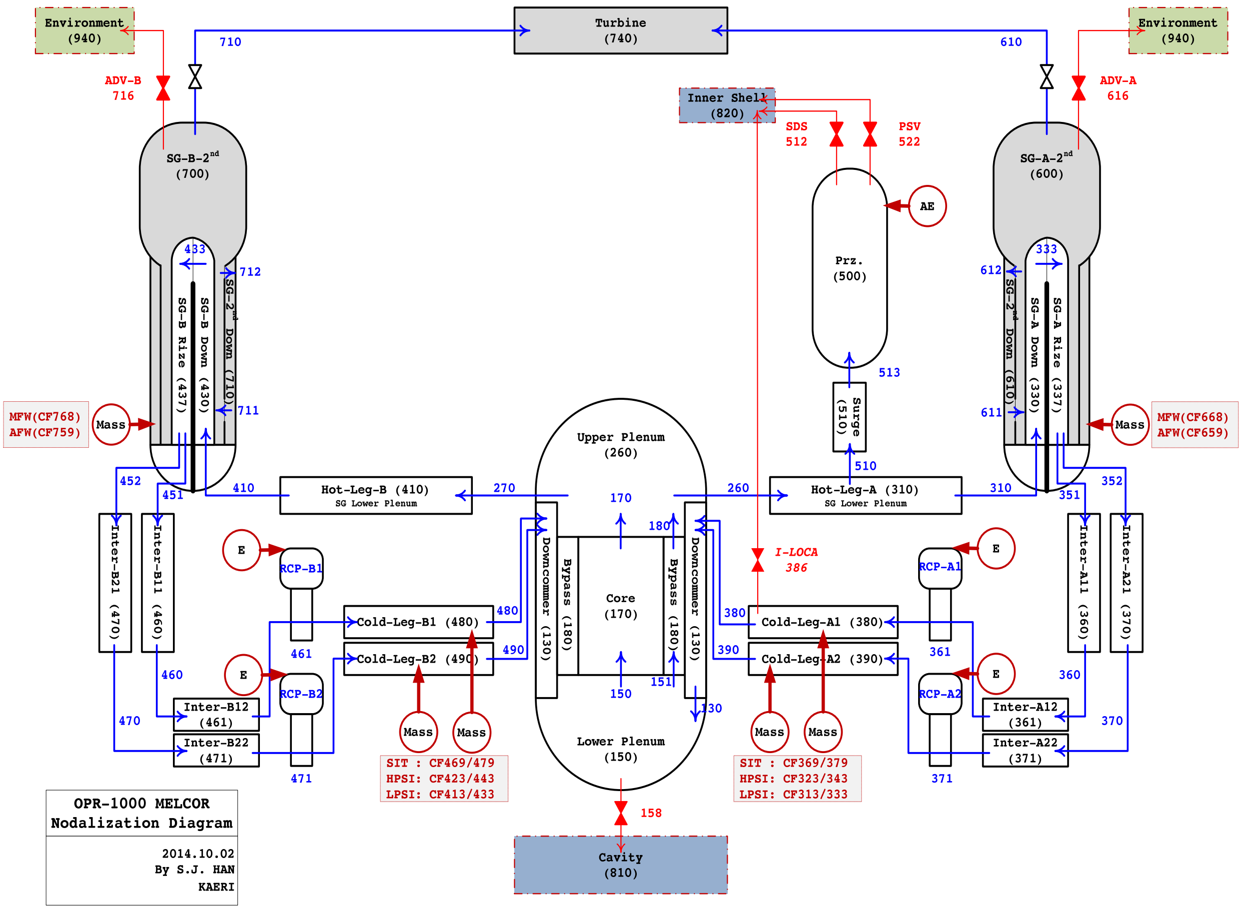


Fig. 1 MELCOR CVH/FL nodalization diagram for OPR-1000, typical Korean PWR

## 1. 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: 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, the assessment of radiological source term effects on the environment has been a revitalized key concern of nuclear safety.

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.

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.

## 2. MELCOR MODELING FOR SBO ACCIDENT

### 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.

The elevations of control volumes are set from reference level of hot leg centerline (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 (core) 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.

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 |

### 2.2 SBO Sequence

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 |

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 reactor coolant system (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 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 is available. The water is automatically injected into the RCS when RCS pressure drops below 4 MPa.

Table 3 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 occurs during from 2.35 to 4.16 h. Finally failures of reactor vessel lower head penetrations occur during about 1 hours 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, $H_2$, $H_2O$ generates 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 which were released from fuel transport and deposit 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.

Table 3. Accident Progression of Given Scenario

| event | hr |
|---|---|
| SBO occurs, Reactor trip, MFW trip, MSIV closure, AFW trip | 0.00 |
| SG-1, SG-2 dryout | 1.59 |
| Core uncover starts | 2.06 |
| Cladding gap release | 2.35 |
| Core support plate failure | 3.77 |
| Reactor vessel lower head (penetration) failure. | 4.16 |
| Core debris ejection to cavity | 4.16 – 5.1 |
| Containment Failure (P > 7 MPa) | 43 |
| Simulation end time | 70 |

## 3. RESULTS AND DISCUSSION

### 3.1 Thermal Hydraulic Behaviors

#### Liquid flow through PORV

Fig.2 shows liquid mass flow rate through pressurizer PORV to containment. Up to 2 hours, the mass flow rate is about 200 kg/s from the RCS to containment when the PORV opens. From 2 to 4 hours, the mass flow rate is less than 100 kg/s when the PORV opens. .

Fig.3 shows integrated liquid flow mass from RCS to containment through pressurizer PORV. Total, 195 of total RCS water inventory was lost from RCS to containment through pressurizer up to 4 h, at which time the reactor vessel lower head failure occurs.

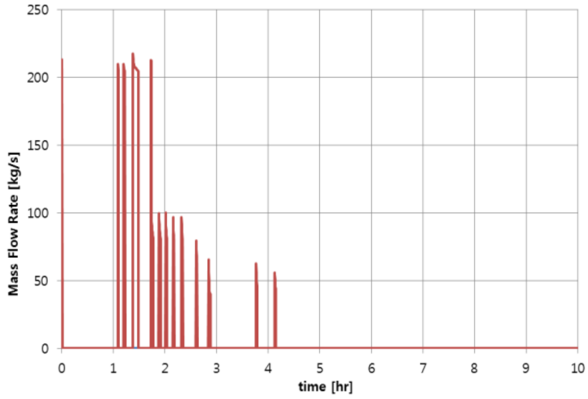


Fig.2 Mass flowrate through pressurizer PORV

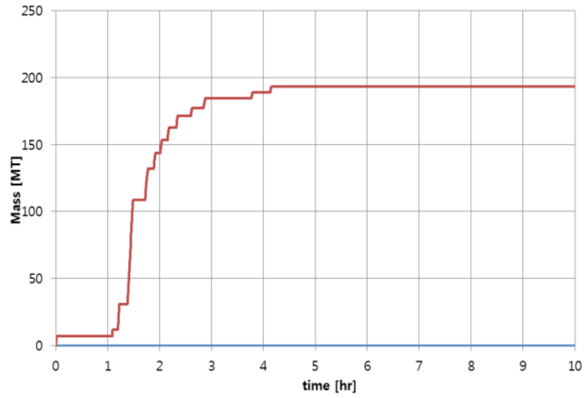


Fig.3 Integrated mass flow through pressuizer PORV

#### Thermal Hydraulic Behaviors in RCS

Fig.4 shows RPV water level transients. Core uncover (TAF) occurs at 2 h due to the reactor coolant discharge to containment. Reactor vessel water level reaches BAF (Bottom of active fuel) level reaches at 2.7 h. Core materials slumping down to lower plenum from 4 to 6 h.

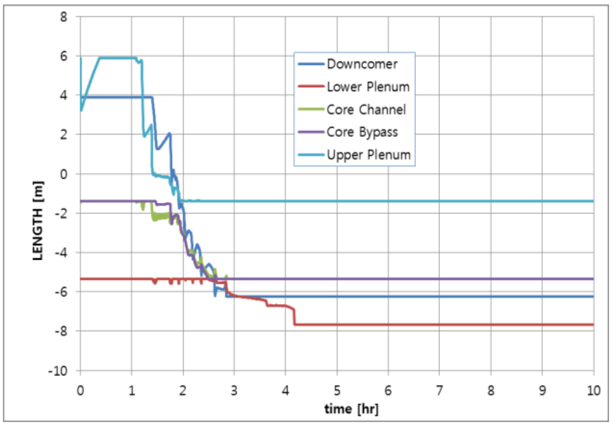


Fig.4 RCS Water levels at each RPV control volume

Fig.5 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 at the cavity due to MCCI is larger than the amount of hydrogen generated in RPV due to 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.

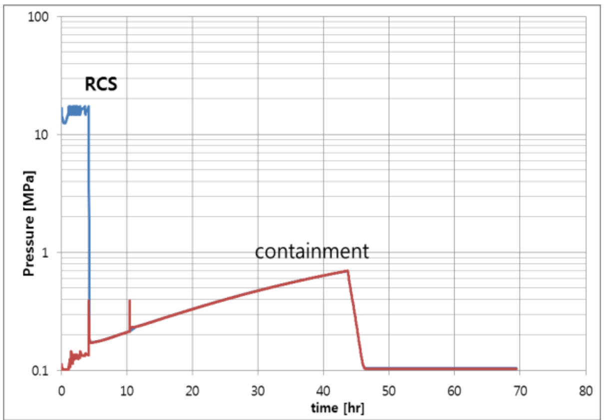


Fig.5 RCS and Containment Pressure

Fig.6 shows liquid temperature of RPV control volumes. Liquid temperature in core channel starts to increase at 4 h. It reaches above 2500 C at about 4 h, when reactor vessel failure occurs. Cladding oxidation, fuel degradation, and relocation occur from 4 to 5 h.

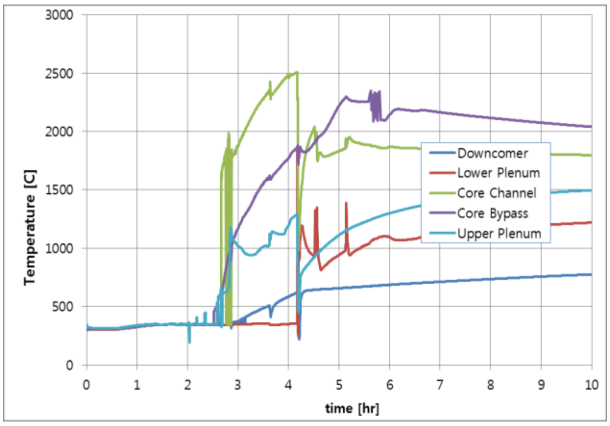


Fig.6 Liquid Temperature of RPV Control volumes

### 3.2 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 2.5 h and ends at 5 h when vessel failure occurs. Fuel cladding failure, support structure failure and core material relocation occurs during this time.

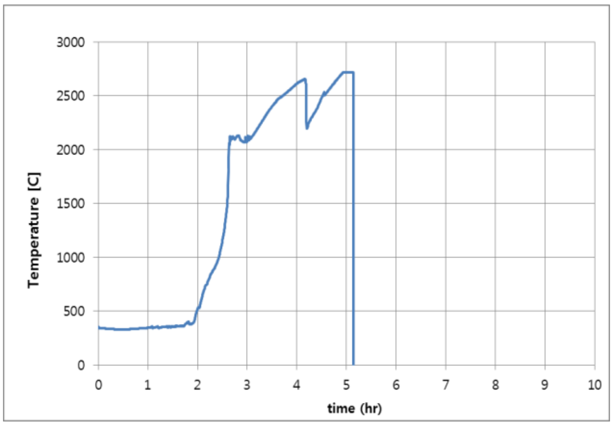


Fig.7 Maximum cladding temperature

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 2.5 h. Core materials ejected to cavity from 4 h to 5 h. Core region is modeled as 7 concentric radial rings.

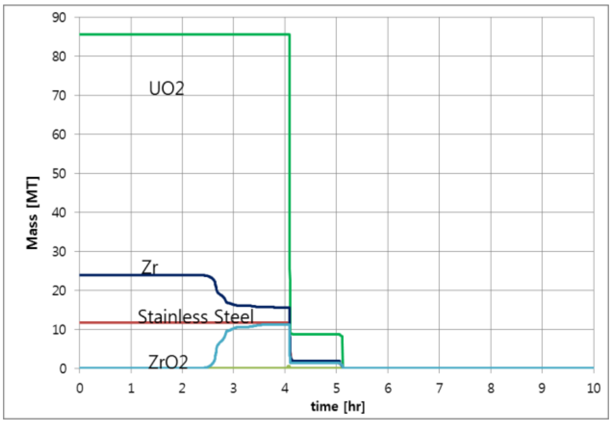

Fig.8 Mass changes of core materials

Fig.9 shows decay heat distribution in core and cavity. In Fig.9, the difference of power between core decay power 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 both 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

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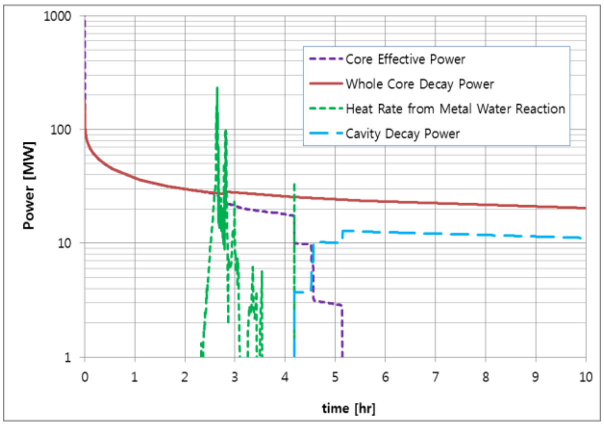

Fig.9 Decay heat distributions

Figure 10 shows hydrogen generation from core. Total 380 kg of hydrogen generated from metal water reaction. Among them, 360 kg of hydrogen generated from zircaloy and 30 kg of hydrogen generated from stainless steel.

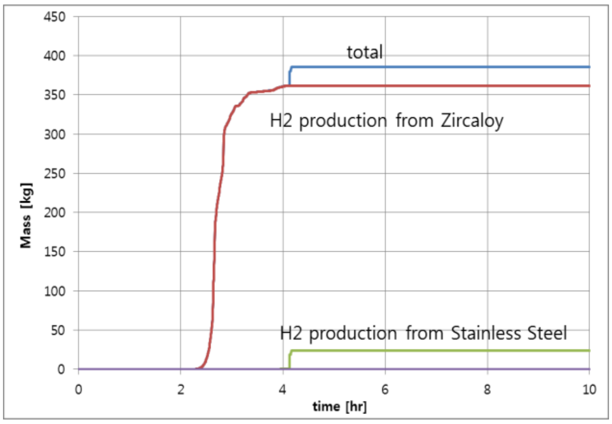

Fig. 10 hydrogen generation from core

### 3.3 Source Term Behaviors in RCS and Containment

#### Fission Product Transport Calculation Scheme in MELCOR Code

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 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 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. Particle diffusion is generally considered to be a relatively unimportant deposition process.

**Fission Product Release from Core**

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

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 |

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 (Te) and antimony (Sn), and 0.0001% for barium (Ba) and strontium (Sr). However, this emission is very small in comparison with the melt release. The amount of gap release is much smaller than the amount of 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.

**Cavity Release from MCCI**

Corium relocated to cavity will interact with basemat concrete on the cavity floor. During this MCCI reaction, radioactive and non-radioactive gases and aerosols will be released. Typical non condensable gases released from MCCI are H2, H2O (steam), CO and CO2. This gases increase containment pressure. Zirconium and stainless steel, which are not yet oxidized in the reactor vessel, will have metal-water reaction (MWR) again with this H2O (steam). Volatile fission products deposited on the RCS walls will be escape from the RCS to the containment atmosphere and then will be released to the environment eventually.

**Environmental Source Terms**

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:

Fig.11 and Fig.12 show the release fractions of Cesium and Iodine to RCS, containment, and environment.

Table 5 shows Cs and I deposited fractions (%) at 70 h in various compartments of plant. Containment failure occurs at 43 h due to the containment pressure reaching 0.7 Mpa. Containment failure pressure of 0.7 MPa ia assumed.

In both cesium and iodine, 100% of initial core inventory is released from fuel. In cesium case, 77% is deposited in the RCS, 21% is deposited in the containment, and 2% is released to the environment. In iodine case, 7% is deposited in the RCS, 82% is deposited in the containment, and 11% is released to the environment.

Cesium and iodine, which start to release from fuel at 2.5 h are deposited on the RCS walls. Before reactor vessel failure occurs at 4 h, about 80% are stayed in RCS and about 20% are released to containment. This behavior is applicable the same way in both of cesium and iodine up to 4 h. After vessel failure occurs at 4 h, most (80%) of the cesium retained in the RCS. However, most (80%) of iodine release to the containment, so that only 20% of iodine stayed in the RCS. Sudden opening between two rooms will make the fission product vapors to evaporate. Evaporation rate of iodine is much larger than that of cesium. This phenomenon will also be applicable to the timing of containment failure at 43 h.

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. 13 shows environmental release fractions of RN classes. This result can be used for off-site consequence analysis for emergency planning and preparedness.

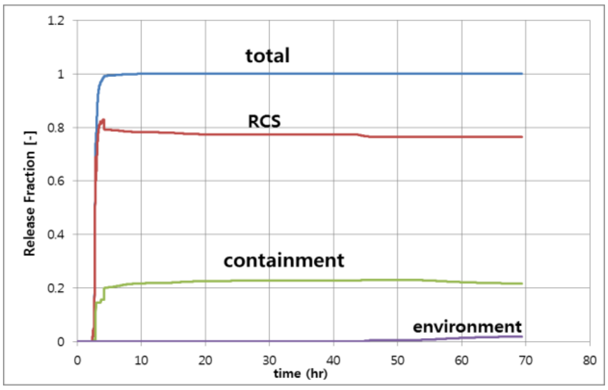


Fig.11 Release fractions of Cesium

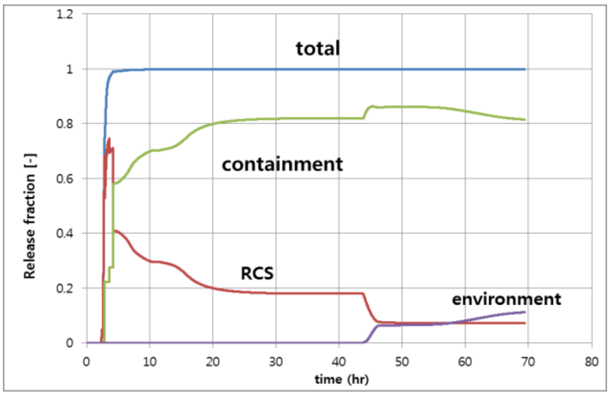


Fig.12 Release fractions of Iodine

Table 5. Cs and I Released Fractions at 70 hr

| Deposited place | Cesium | Iodine |
|---|---|---|
| Deposited in RCS | 0.77 | 0.07 |
| Deposited in Containment | 0.21 | 0.82 |
| Released to Environment | 0.02 | 0.11 |
| Total release from Fuel | 1.0 | 1.0 |

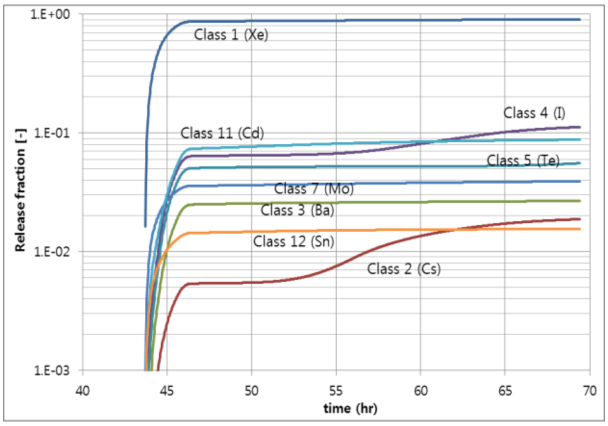


Fig.13 Environmental realease fraction of each RN class

## 4. CONCLUDING REMARK

In the MELCOR simulation for SBO of OPR-1000, core damage starts at 2.5 h and reactor vessel lower head (penetration) failure occurs at 4.1 h. There is no fission product release to environment yet, because containment pressure did not reach to containment failure pressure yet.

In both cesium and iodine, 100% of initial core inventory is released from fuel. In cesium case, 77% is deposited in the RCS, 21% is deposited in the containment, and 2% is released to the environment. In iodine case, 7% is deposited in the RCS, 82% is deposited in the containment, and 11% is released to the environment.

Fukushima source term released to environment is estimated in the UNSCEAR-2013 report. 2 ~ 8 % of iodine is release to environment while 1 ~ 3 % of cesium is release to environment from the Fukushima Daiichi Accident at March 2011.

Very similar results obtained from this sample study.

| Nuclide | SBO in this study | Fukushima Daiichi, 2011 |
|---|---|---|
| I | 11% | 2-8% |
| Cs | 2% | 1-3% |

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## ACRONYMS

| | |
|---|---|
| AC | Alternate Current |
| ADV | Atmospheric dump valve |
| AFW | Auxiliary Feedwater |
| BAF | Bottom of Active Core |
| BWR | Boiling Water Reactor |
| CSP | Core Support Plate |
| DC | Direct Current |
| HPSI | High pressure safety injection systems |
| ISLOCA | Interfacing Systems LOCA |
| KEPRI | Korea Electric Power Research Institute |
| KHNP | Korea Hydro and Nuclear Power Co. |
| KAERI | Korea Atomic Energy Research Institute |
| LPSI | low pressure safety injection |
| LOCA | Loss of Coolant Accident |
| LWR | Light Water Reactor |
| MCCI | molten-core-concrete-interaction |
| MFW | Main feedwater |
| MSIV | Main steam line isolation |
| MD | Motor driven |
| NRC | Nuclear Regulatory Commission, USA |
| OPR | Optimized Power Reactor |
| PWR | Pressurized Water Reactor |
| RCP | Reactor Coolant Pump |
| RCS | Reactor Coolant System |
| RHRS | Residual heat removal system |
| RPV | Reactor Pressure Vessel |
| SBO | Station Blackout |
| SCS | Shutdown cooling system |
| SG | Steam generator |
| SGTR | Steam Generator Tube Rupture |
| SIT | Safety injection tank |
| SNL | Sandia National Laboratories, USA |
| SOARCA | State of Art Reactor Consequence Analysis |
| TAF | Top of Active Core |
| TD | Turbine driven |
| TMI | Three Mile Island |
| UNSCEAR | United Nations Scientific Committee on the Effects of Atomic Radiation |
| WHO | World Health Organization |