Dynamic and Interactive Approach to Level 2 PRA Using Continuous Markov Process with Monte Carlo Method

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The current approach to Level 2 probabilistic risk assessment (PRA) using the conventional event-tree(ET)/fault-tree(FT) methodology requires pre-specifications of event order occurrence and component failure probabilities which may vary significantly in the presence of uncertainties. The possible scenarios, which results in massive fission products release, depends on the event order occurrence and the instantaneous plant state. Thus, manual preparation of input data to evaluate the possible scenarios arising from these uncertainties may also lead to errors from faulty/incomplete input preparation. In the present study, a new methodology is proposed to quantify the level 2 PRA in which the accident progression scenarios are dynamic and interactive with the instantaneous plant state and related phenomena. The accident progression is treated as a continuous Markov process and the transition probabilities are evaluated based on the computation of plant system thermal-hydraulic dynamics. A Monte Carlo method is used to obtain the resultant probability of the radioactive material release scenarios. The methodology is applied to the protected loss of heat sink accident scenario of the level 2 PRA of a generation IV fast reactor. It is confirmed that the new approach consistently describes the time-transient of the phenomena and the system state during a severe accident. It is a dynamic and interactive approach that couples the source term behavior with system thermal hydraulics and structural response of physical barriers seamlessly

I. Introduction

The level 2 probabilistic risk assessment (PRA) of a nuclear power plant deals with accident progression scenarios that lead to the release of radioactive materials to the environment following the reactor core damage and physical barrier failures. Therefore, consequences of the accident depend on core damage conditions, containment failure modes, and source term categories. The process of fission product release from the reactor core to the environment is needed to be evaluated in a mechanistic and probabilistic manner. For this purpose, ET and FT methodology are used for the quantification of the risk of events developing a process in the level 2 PRA. These methodologies require the pre-specifications of event order occurrence and component failure probabilities which may vary significantly in the presence of uncertainties. Thus, it is easily foreseen that the modes of the core damage and the structural boundary failure varies considerably, and the chronological sequence of the failure is indefinite. This fact leads us to the conclusion that the conventional event tree approach cannot apply to cover the whole scenarios of the level 2 PRA.

In the previous studies [1-3], a new methodology based on a Continuous Markov chain Monte Carlo (CMMC) method was proposed to quantify an uncertainty and time-dependent accident scenario that cannot be modeled by using the conventional ET method. The CMMC method evaluates the current system state probability sequentially using the current plant state. The current system state probability decides the plant state on the next step. Thus, the conventional ET is not necessary to determine a branch probability. By coupling thermal-hydraulic and mechanical system response models with the CMMC method, event probabilities such as a failure of structure and component by excessive thermal load was able to evaluate. The CMMC method was implemented for the scenario quantifications of the protected loss of heat sink (PLOHS) accident of the sodium-cooled reactor.

In this study, source term analysis is carried out under the PLHOS accident to validate the applicability of the CMMC method to the level 2 PRA research.

II. Coupling method for accident scenario quantification

II.A. CMMC method

The CMMC method is based on the MCMC (Markov Chain Monte Carlo) method [2-3]. According to the Markov model, any subsequent state depends only on the state preceding it. Thus, the accident progression is consistently described as the Markov process. We define that the plant state on the next time step is only dependent on the current plant state. Although failure probabilities per unit of time for each component depends on the previous data (For example, temperature and pressure), we adopt them as cumulative values which are based on the previous data at each time. Furthermore, to determine the plant conditions, we compare the state transition probability (branch probability for the ET) and a random number which is created by the Monte Carlo method. To determine the state transition probability thermal-hydraulic parameters such as temperature and pressure are required. These parameters can be obtained by a thermal-hydraulic analysis code. Then the subsequent plant state is decided by the current state transition probability. By iterating these process until the termination of the analysis, a unique accident scenario is obtained (As shown in Fig.1). A large number of independent (particular) accident scenarios can be obtained by this procedure because each scenario depends on the state transition probability and random number by the Monte Carlo method. This approach of coupling the MCMC method with the thermal-hydraulic code is the Continuous Markov chain and Monte Carlo (CMMC) method. The CMMC method is applied to PLOHS (Protected Loss of Heat Sink) accident of a sodium-cooled fast reactor to evaluate the fission products release from the reactor core.

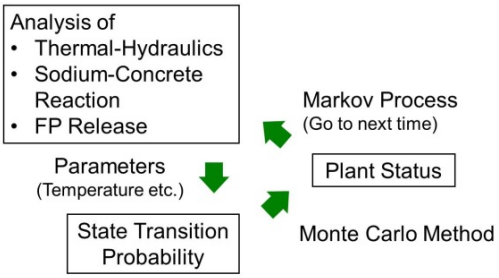


Fig.1 Concept of the CMMC coupling method

II.B. PLHOS accident

A PLOHS (Protected Loss of Heat Sink) accident is one of the most severe accidents in Sodium-cooled Fast Reactor (SFR) [4]. In SFR plant, the Auxiliary Cooling System (ACS) remove decay heat from the reactor core. Thus, ACS failure after the reactor scram can bring the following sequence. 1) Loss of heat removal sources after the scram, 2) temperature and pressure increase gradually in the reactor core and coolant boundary structures, 3) the fuel damage and the coolant boundary failure occur by high temperature and pressure. 4) fission products (FP) are released to the outside with the coolant leakage. In PLOHS accidents, there are two major FP release scenarios; a containment vessel failure (CV-Failure) and a containment vessel bypass (CV-bypass) scenario. The CV-Failure scenario takes places when both the primary loop and the containment vessel are failed. On the other hand, the CV-bypass scenarios occur when the both IHX and secondary-loop are failed. Thus, for the CV-bypass scenarios, the coolant which contains fission products can leak to the outside of the plant without the CV-Failure.
In this study, we add an analysis model for fission products transport behavior to the Shinzaki's meta-model [3] to evaluate the fission product release from the severe accident.

III. Numerical Analysis Model for Fission Products Transport Behavior

Fission products release path depends on the accident scenario. The amount and timing of released fission products and inventory in each component vary according to the occurrence time for fuel failure, fission products leakage path and containment vessel failure. Also, FPs are variable since its behavior is influenced by radioactive decay of nuclides and absorption in the wall of plant components. Therefore, the thermal-hydraulic analysis code and FP transport analysis code, which can compute transient plant state consistently, is needed.

III. A. Meta-model for PLOHS Accident Analysis

In the previous study, Shinzaki proposed a meta-model using the CMMC method coupled with NALAP-II code for the quantification of the level 2 PRA [5]. The NALAP-II code has been developed by JNES (currently NRC), Japan. The code can evaluate thermal-hydraulic properties such as temperature and pressure for each component (primary loop and secondary loop) of SFR (Fig.2). The model calculates failure probability per unit of time for each component, leakage of coolant and FP release at each time in each sample. Each judgment to decide failure is decided by a Monte Carlo method.

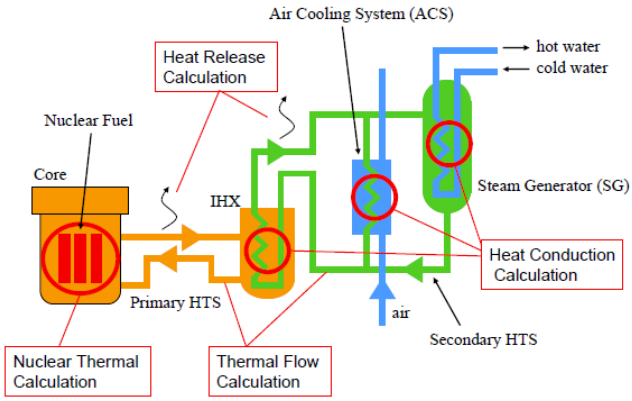


Fig.2 Outline of NALAP code

III.B. Fission Product Transport Analysis Code

In the former model, only one nuclide was considered to analyze fission product behavior for the various accident scenarios. However, for the adequate source term analysis, kinetics for fission products transport behavior should be considered. Thus, in this study, we propose a kinetic model for fission product release. As mentioned above, the postulated scenarios which bring FP release are the CV-bypass and CV-failure. In the case of the CV-bypass, the amount of released the FPs is evaluated by considering the mass of radioactive nuclides in coolant liquid sodium. For the CV-failure scenario, the amount of released FP is calculated by an internal release rate of nuclides which deposit in the containment vessel. A deposition of nuclides on a wall of components in the reactor and radioactive decays are considered in the model.

III.B.1 Evaluation model for nuclide in coolant

The amount of nuclides in the coolant is evaluated by considering terms of transport, disposition, consumption and generation by radioactive decay.

$$\rho_j V_j \frac{dC_{pk}}{dt} = W_j\left(C_{uk} - C_{pk}\right) - \Gamma_{jk} A_j \rho_j C_{pk} - \rho_j V_j \sum^{m} \lambda_l C_{pl} \qquad (1)$$

where
ρ : density [kg/m^3], V : volume [m^3], A: contact area with a component wall [m^2], λ: deposition rate coefficient [m/sec]
subscript k,l: nuclide, m: parent nuclide, j: evaluation node, u: upper node

(Deposition rate for non-volatile nuclide)
Also, the deposition rate coefficient and the mass transfer coefficient are defined as follow

$$\Gamma_{jk} = K_{L_j} S_k \qquad (2)$$

where
Γ_{jk} : mass transfer coefficient on node j in coolant [m/sec]
S_k : sticking number for nuclide k [-]

$$K_{L_j} = 0.023 \cdot Re^{0.83} \cdot Sc^{0.33} \cdot D/d \qquad (3)$$

Where Re, Sc are the Reynolds number [-] and Schmidt number [-], respectively. D is diffusion coefficient and d is radius [m]

(Deposition rate for volatile nuclide)
The deposition rate for volatile nuclide is defined as below

$$\Gamma_{jk} = K_k^e \lambda_k \quad (4)$$

where
K_k^e: partition coefficient for nuclide k [m]
λ_k : Decay constant [1/sec]

For each non-volatile nuclide, the sticking numbers are prepared as input data.

III.B.2 Evaluation model for wall deposition

Nuclides in reactor coolant are deposited on the surface of the reactor component wall by absorption. The deposition mechanism is based on a concentration diffusion from turbulent region to the reactor wall component through a laminar region. Therefore, the amount of deposited nuclides is calculated by using the Sticking number and Partition coefficient.

(Wall deposition model for non-volatile nuclide)

Deposition of non-volatile nuclides is an irreversible procedure, once deposited nuclides are hard to be desorbed. Thus, the amount of deposited nuclide per unit time can be calculated as follow;

$$j_k = S_k K_c C_K \quad (5)$$

Where,

j_k: deposition rate of nuclide k [atoms/m^2/sec]
S_k: sticking coefficient of nuclide k [-]
K_c: molecular transfer coefficient of [atoms/m^2/sec]
C_k: concentration of nuclide k in coolant [atoms/m^3]

(Wall deposition model for volatile nuclide)

For volatile nuclides, the desorption of nuclide is not negligible. Thus, we assume that the deposition and desorption on the wall are in an equilibrium state. A partition coefficient is used to calculate the amount of deposited nuclide as follow;

$$K_k^e = \frac{C_k^w}{C_k} \quad (6)$$

where,
K_k^e: partition coefficient for nuclide k [m]
C_k^w: deposition concentration for nuclide k [atoms/m^2]
C_k : concentration for nuclide k in coolant [atoms/m^3]

The partition coefficient and the stocking coefficient are referred to experiment data.

III.B.3 Model for radioactive nuclides in fuel

Before the fuel failure, the amount of radioactive nuclides in fuel is changed by radioactive decay chains. When the fuel fails, a discharge of nuclides to liquid sodium is added. In this study, the discharge of radioactive nuclides to the coolant is assumed to stop when coolant inventory in the containment vessel becomes a half of the containment volume.

$$\frac{dX_{jk}}{dt} = -K_k X_{jk} - \lambda_k X_{jk} + \sum^{m} \lambda_l X_{jl} \quad (7)$$

where,
X_{jk}: amount of FP in the fuel [mol]
K_k : release velocity for nuclide k [fraction/sec]

The discharge velocity is defined for each nuclide as follow

$$K_k = A_k \cdot exp(B_k \cdot T_j) \quad (8)$$

The coefficient $\boldsymbol{A_k}$ and $\boldsymbol{B_k}$ are referred to the modified NUREG model[6] which is based on the NUREG-0722[7].

III.B.4 Model for release rate from the containment vessel

The amount of release rate from the containment vessel is decided by a function of leakage rate of coolant from the boundary of reactor coolant.

$$W_{CVout} = \alpha_{CVleak} m_{CVleak} \quad (9)$$

where

W_{CVout}: release rate from the containment vessel [mol/s]
α_{CVleak}: coefficient for the containment vessel release rate [s^{-1}]
m_{CVleak}: release rate for radioactive nuclide from reactor coolant boundary (remain amount in containment vessel) [mol]

III.C. Embedment of FP transport model to thermal hydraulics code

The FP transport model is embedded in the Shinzaki's thermal hydraulic code as shown in Fig.3. At first, plant state is calculated by the thermal hydraulic code at each time step, then based on the mass flow rate of the coolant and temperature, the fission product transport calculation is carried out.

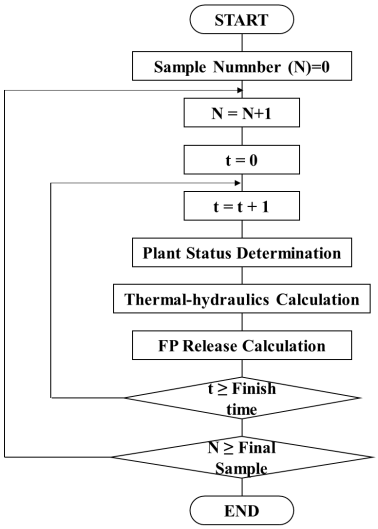


Fig. 3 Fission Product Transport Analysis Model Algorithm

The validation of the FP transport model was evaluated by considering the following 4 points.

(a) Mass conservation of FPs in steady state flow
(b) Mass conservation of FPs in unsteady state flow
(c) Mass conservation of FPs considering radioactive decay term
(d) Mass conservation of FPs considering deposition term

IV. Fission Product Behavior and Source Term Analysis for PLOHS Accident

IV.A Analytical Model and Conditions

In this study, a loop type sodium-cooled fast reactor which has two loops (A and B) are used for postulated system. Each loop has IHX (Internal Heat Exchanger), steam generator, auxiliary cooling system and decay heat removal system. For Japanese prototype sodium-cooled fast reactor, Monju has 3-loop systems. For the sake of convenience, however, 2-loop system which has the same the coolant inventory with the 3-loop system is used. As broken location, the reactor vessel lower plenum, the primary system pipe (a total of four locations that is hot leg and the cold leg for two loops), secondary loop pipe (a total of

eight locations in the two loops, those are hot leg, one place containment vessel inside and outside for each of the cold leg), SG and the containment vessel are selected.

For 42 nuclides, initial inventories are prepared as input data. Figure 4 shows some of the decay chains of 42 nuclides. Decay chains marked with a red square are considered in the analysis. The others are neglected since they have a negligible very short half-life. As a result, total 78 nuclides are used for the source term analysis. For each nuclide, the sticking coefficient, partition coefficient and release velocity coefficient are given as shown in Table 1. Table 2 shows the analytical conditions. The meta model output at 20,000 seconds after the success of reactor scram, is used as input data. In the case of the PLHOS accident, the reactor coolant boundary is broken by a high-temperature creep, we assume that when the outlet temperature of the reactor core is below 550℃ during normal operation, a rupture of the coolant boundary does not occur. Our calculation starts at 20,000 seconds after the reactor scram is successful.

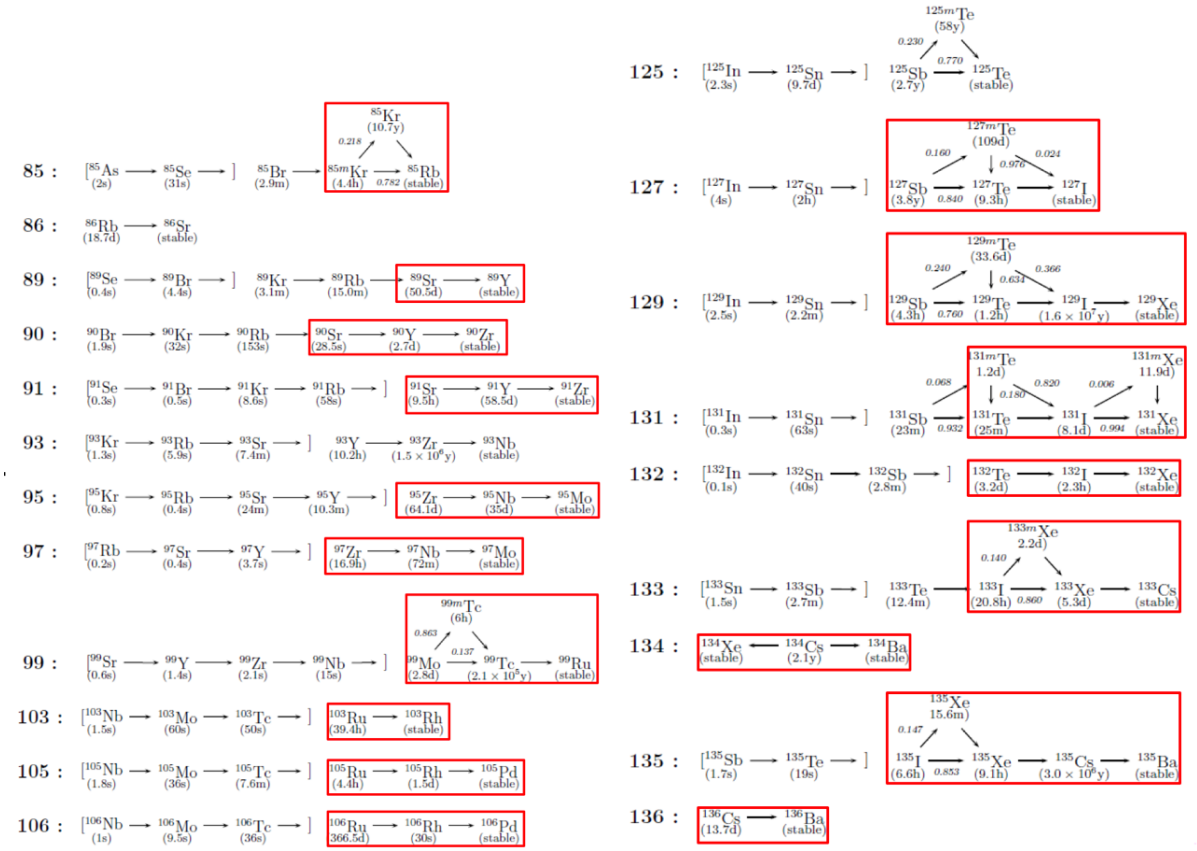


Fig.4 Postulated Decay chain for nuclides

Table1. Input data for 78 nuclides

No.	Nuclide	Decay Const.	Init. Inventory [kg]	No.	Nuclide	Decay Const.	Init. Inventory [kg]	No.	Nuclide	Decay Const.	Init. Inventory [kg]	No.	Nuclide	Decay Const.	Init. Inventory [kg]
1	Kr85	4.30E-05	5.76E-04	21	Tc99m	3.20E-05	7.49E-03	41	Te131m	6.42E-06	5.25E-03	61	Cs136	6.17E-07	3.54E-02
2	Kr85m	2.05E-09	3.43E-01	22	Tc99	1.03E-13	0.00E+00	42	Te131	4.62E-04	0.00E+00	62	Ba136	0.00E+00	0.00E+00
3	Rb85	0.00E+00	0.00E+00	23	Ru99	0.00E+00	0.00E+00	43	I131	9.98E-07	2.39E-01	63	Cs137	7.30E-05	2.60E+01
4	Rb86	4.30E-07	2.05E-03	24	Ru103	2.03E-07	1.57E+00	44	Xe131m	6.69E-07	2.45E-03	64	Ba131	0.00E+00	0.00E+00
5	Sr86	0.00E+00	0.00E+00	25	Rh103	0.00E+00	0.00E+00	45	Xe131	0.00E+00	0.00E+00	65	Ba140	6.29E-07	5.64E-01
6	Sr89	1.59E-07	0.00E+00	26	Ru105	4.34E-05	5.64E-03	46	Te132	2.50E-06	1.29E-01	66	La140	4.78E-06	7.48E-02
7	Y89	0.00E+00	0.00E+00	27	Rh105	5.45E-06	4.48E-02	47	I132	8.43E-05	3.84E-03	67	Ce140	0.00E+00	0.00E+00
8	Sr90	7.71E-10	6.35E+00	28	Pd105	0.00E+00	0.00E+00	48	Xe132	0.00E+00	0.00E+00	68	Ce141	2.48E-07	1.46E+00
9	Y90	3.00E-06	2.00E-03	29	Ru106	2.16E-08	6.59E+00	49	I133	9.26E-06	4.45E-02	69	Pr141	0.00E+00	0.00E+00
10	Zr90	0.00E+00	0.00E+00	30	Rh106	2.28E-02	0.00E+00	50	Xe133m	3.60E-06	1.76E-02	70	Ce143	5.83E-06	5.00E-02
11	Sr91	2.03E-05	5.73E-03	31	Pd106	0.00E+00	0.00E+00	51	Xe133	1.52E-06	2.75E-01	71	Pr143	5.91E-07	0.00E+00
12	Y91	1.37E-07	8.01E-01	32	Sb127	2.09E-06	1.35E-02	52	Cs133	0.00E+00	0.00E+00	72	Nd143	0.00E+00	0.00E+00
13	Zr91	0.00E+00	0.00E+00	33	Te127m	7.36E-08	5.37E-02	53	Xe134	0.00E+00	0.00E+00	73	Ce144	2.82E-08	6.81E+00
14	Zr95	1.25E-07	1.67E+00	34	Te127	2.06E-05	1.34E-03	54	Cs134	1.07E-08	4.52E-01	74	Pr144	6.69E-04	0.00E+00
15	Nb95	2.29E-07	8.27E-01	35	I127	0.00E+00	0.00E+00	55	Ba134	0.00E+00	0.00E+00	75	Nd144	-1.05E-23	0.00E+00
16	Mo95	0.00E+00	0.00E+00	36	Sb129	4.46E-05	1.79E-03	56	I135	2.92E-05	1.41E-02	76	Nd147	7.25E-07	2.12E-01
17	Zr97	1.14E-05	2.11E-02	37	Te129m	2.39E-07	8.51E-02	57	Xe135m	7.55E-04	1.27E-04	77	Pm147	8.37E-09	0.00E+00
18	Nb97	1.00E-04	0.00E+00	38	Te129m	1.66E-04	4.53E-04	58	Xe135m	2.10E-05	2.18E-02	78	Sm147	-2.05E-19	0.00E+00
19	Mo97	0.00E+00	0.00E+00	39	I129	1.40E-15	0.00E+00	59	Cs135	9.55E-15	0.00E+00				
20	Mo99	2.92E-06	9.53E-02	40	Xe129	0.00E+00	0.00E+00	60	Ba135	0.00E+00	0.00E+00				

Table 2. Analytical conditions for FP release analysis

Parameter	Value
Total sample number	100,000
Analysis starting time	20,000 sec after a reactor scram
Analysis finish time	220,000 sec after a reactor scram
Heat value of reactor core	Quote from a data table of decay heat
Plant status	PLOHS (heat insulation)
Heat removal sources	No heat removal sources
Heat release from components	No heat release

IV.B Fission Products Release Analysis

First of all, release rate for 78 nuclides is evaluated using a single sample. Figure 5 shows a change of the ratio of FP inventory as the accident proceeds. In this calculation, the accident progression happens in the following order. At 60,000 second, the fuel failure takes place. Secondary loop failure (CV-bypass) which brings FPs release to the environment occurs at 130,000 sec. At 190,000 sec, the containment vessel fails and the FPs leak into the environment.

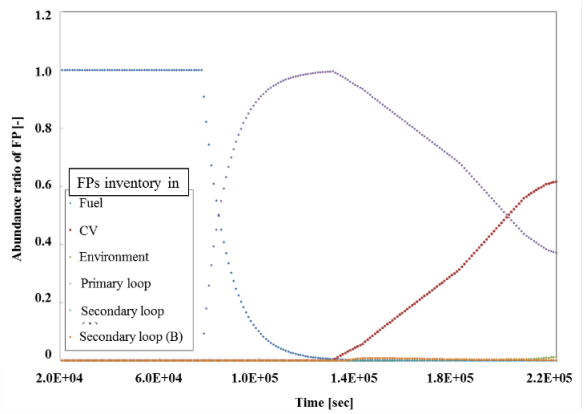
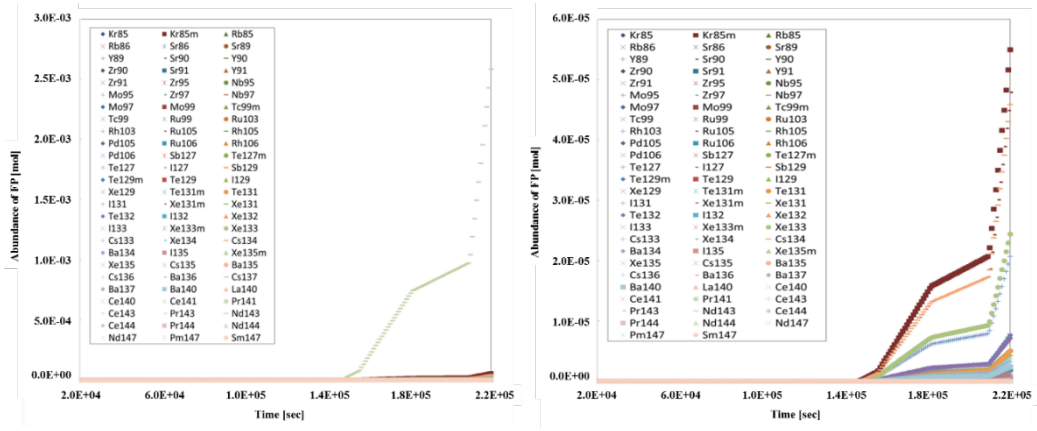


Fig.5 Fission products inventory in each component

Figure 6 and 7 shows the analytical results for 78 nuclides. Cs137, Kr85, Cs134, Sr90, Xe133 and I131 are dominant discharged nuclides. Since the release rate is calculated by considering the FP behaviors (the initial inventory, deposition, desorption, mass transfer velocity and decay), the above nuclides can be regarded as dominant FPs which release to the environment. Cs137 has the highest release rate since initial inventory for Cs137 is larger than other nuclides.



(a) Including Cs137 (Whole) (b) Excluding Cs137

Fig. 6 Discharge rate for 78 nuclides to environment

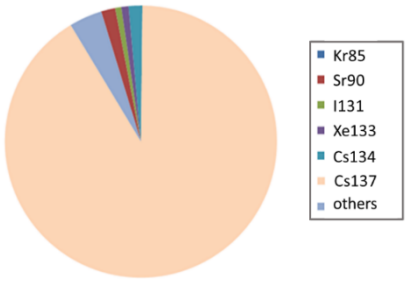


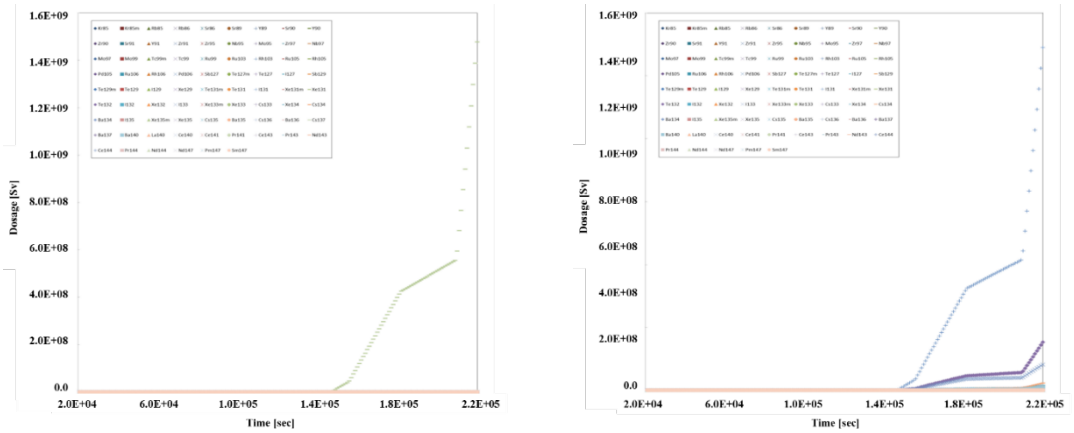
Fig.7 Ratio of released FPs to the environment after 220000 sec

To evaluate the impact of FPs release to the public, we divert the result to dose evaluation. The equivalent dosage is estimated by considering decay constant and the dose conversion factor for each nuclide. The following equation is used for does conversion.

$$D = \sum_{k} \lambda_k \cdot C_k \cdot DF_k \tag{10}$$

Where,
D : Dosage [Sv]
C_k: Release rate for nuclide k [mol]
λ_k: Decay constant for nuclide k [sec^{-1}]
DF_k: Dose conversion factor [Sv/Bq]

Figure 8 and 9 shows the converted dose for 78 nuclides from the release rate in fig. 6 and 7. The dominant nuclides which have high dosage are Cs137, I131, Te132, I133, Cs134, Cs136.



(a) Including Cs137 (Whole) (b) Excluding Cs137

Fig. 8 Converted dosage from release rate for 78 nuclides to environment

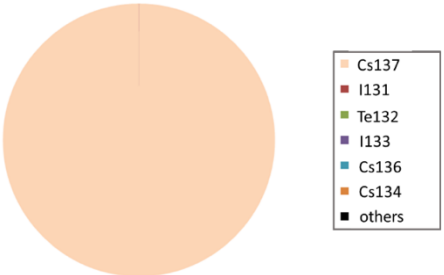


Fig.9 Ratio of converted dosage from FPs release after 220000 sec

As shown above, Cs137, I131, Te132, I133, Cs134, Cs136 have relatively high converted dosage. However, Cs137 is a most dominant source which has high dosage impact to public. From the above results, it is concluded that Cs 137 is the most dominant nuclide which has a high impact on the environment and public.

IV.C Uncertainty Analysis for Accident Scenario

All scenarios generated by the CMMC method are a unique one. Thus, quantification of the accident scenario can be achieved by treating all scenarios statistically. In this study, we carried out analyses using 100 samples and statistical treatment of the results.

First of all, we categorize the total of 78 nuclides into nine groups according to WASH-1400 report as shown in Table 3[8]. The categorization for FPs release is important from the viewpoint of evaluation for dosage to human.

The results of 100 samples analysis also show that Cs137 is dominant nuclide in abundance and dosage (Fig. 10). It seems that the abundance rate and dosage increase exponentially for all nuclides. Because the cumulative abundance and dosage gradually increase along the accident progression. Also as shown in fig.5, the CV failure which leads a large discharge of nuclide to environment take places in latter analysis time.

Figure 11 shows that the aerosol (in which Cs 137 is included) is a dominant state for discharged FPs. Therefore, it is necessary that the impact of the aerosol and gas should be evaluated, respectively. These results have an important role in the level 3 PRA

Table 3. nine groups of categorized nuclides

Group	Name	Nuclide	State
1	Noble Gases	Xe, Kr	Gas
2	Halogens	I, Br	Gas・Aerosol
3	Alkali Metals	Cs, RB	Aerosol
4	Tellurium group	Te, Sb, Se	Aerosol
5	Barium, Strontium	Ba, Sr	Aerosol
6	Noble Metals	Ru, Rh, Pd, Mo, Tc, Co	Aerosol
7	Cerium group	Ce, Pu, Np	Aerosol
8	Lanthanides	La, Zr, Nd, Eu, Nb, Pm, Pr, Sm, Ym, CM, AM	Aerosol

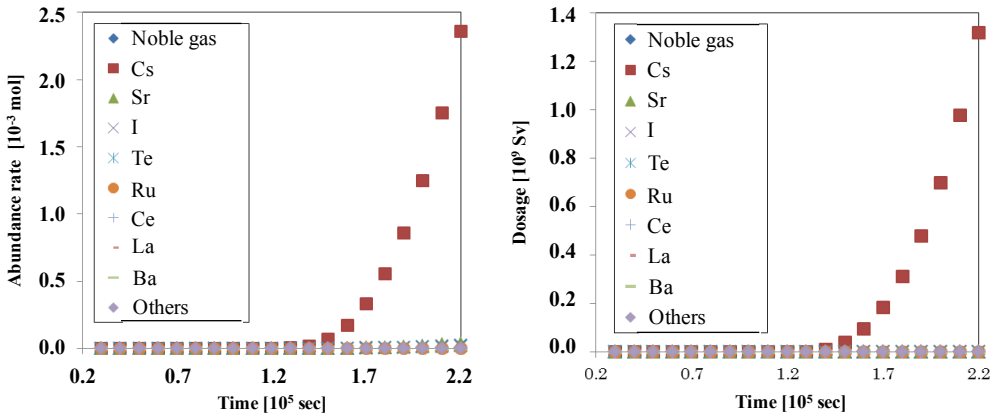


Fig.10 Amount of Discharged Fission Products (Left) and Dosage (Right) according to group

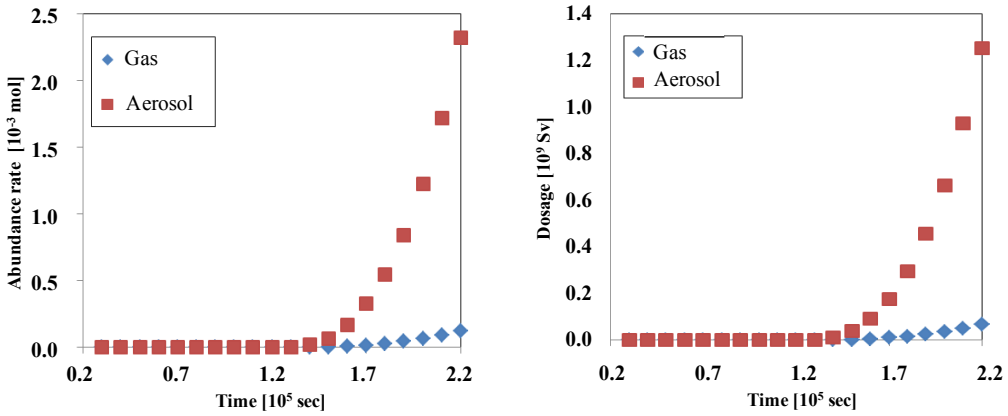


Fig.11 Amount of Discharged Fission Products (Left) and Dosage (Right) according to state

Figure 12 shows that the discharge rate of the FPs released to the environment changes as the accident progression. The two peaks of the discharge rate are existed. It is expected that the former one (about 150,000sec) is caused by the CV bypass event, the later one (about 210,000 sec) is caused by the CV failure event. The result reveals that when the FP discharge to the environment take place and how much the discharge rate will be. This information is important and useful to evaluate the time margin before a large amount of FP release (which exceeds a certain level) takes place. The time margin provides a basis to evaluate how fast the evacuation and accident management actions should be carried out.

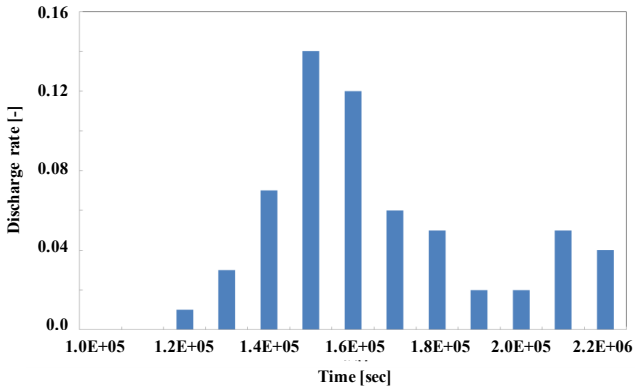


Fig.12 Fission Product Release Rate

Figure 13 shows the converted dosage histogram plotted with total released amount and time. In the case of the PLOHS accident, the CV bypass is a dominant event which brings FP discharges to the environment in an early stage. After 160,000 sec, the CV failure event becomes dominant source for FP release. It is also known that the CV bypass scenario is a medium-scale event which has relatively low frequency and early release timing. On the other hand, the CV failure scenario is a large-scale event which has relatively high frequency and late release timing. From this figure, we can also evaluate how the FP release changes as the event progress. Thus, a mission time to mitigate the accident progress can evaluate based on this result. Also, the importance of equipment and devices for safety activities can be evaluated.

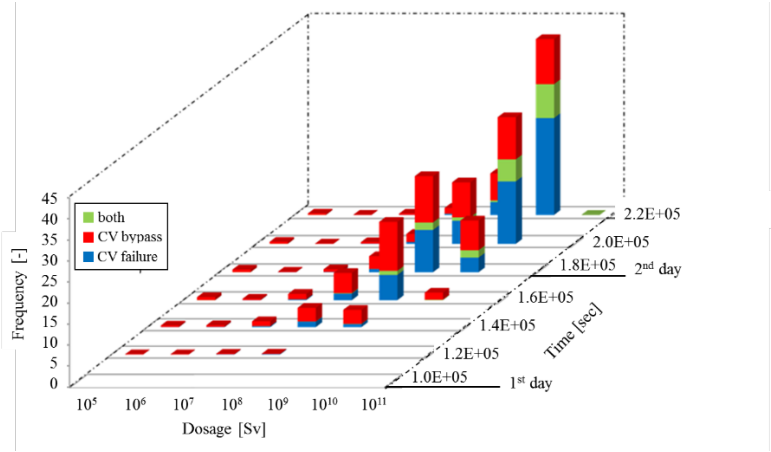


Fig.13 Fission Product Release Histogram (Dosage)

V. Conclusion

The new methodology, which coupled the CMMC method to the thermal-hydraulic analysis code, is proposed for the quantification of accident scenarios which is considered an event progression and a react response. Also, the fission product behavior model which deals with deposition, decay and absorption are proposed and applied to the thermal-hydraulic code. The new methodology is applied to the PLOHS accident of the SFR to evaluate the FP release to the environment. Two accident scenarios (CV bypass event and CV failure event) are estimated as a leakage path for FPs to the environment. As a result of the analysis, it is known that the Cs 137 is dominant nuclides discharged to the environment from the viewpoint of the released amount and dosage. The more fission product released in the aerosol state than in gas state. Also, it is known that the FP release behavior differs according to the failure mode and it can help us to evaluate the time margin for an emergency evacuation and mitigation activities. Through this study, it is shown that the new methodology is a useful way to evaluate the source term analysis considering the event progression and the reactor response in the level2 PRA and provide a basement to evaluate an impact of FPs to public in the level 3 PRA.

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