STUDY ON THE REALISTIC FISSION PRODUCT RELEASE ESTIMATION IN THE SEVERE ACCIDENT ANALYSIS FOR THE SPENT FUEL POOL

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Severe accident analysis of the spent fuel pool (SFP) in an ABWR was conducted using the MAAP5 (Modular Accident Analysis Program, version 5) code. In the MAAP5 code, the total fission product mass in the spent fuel bundles is assumed to be proportionally distributed according to decay heat of each fuel bundle. When the 'Hot' and 'Cold' fuel bundles are defined in the SFP analysis and only 'Hot' fuel bundles are damaged, large amount of fission product, which is included not only Hot fuel bundles but also Cold fuel bundles, was released to the environment in the Hitachi-GE's process. This was too conservative for Level 3 Probabilistic Safety Assessments (PSA). Thus, a modified evaluation method for fission product release mass fraction in the SFP using MAAP5 was examined in this study.

I. INTRODUCTION

Since the Fukushima Daiichi nuclear accident on the 11th of March 2011, the risk of significant radioactive fission product release from the Spent Fuel Pool (SFP) has been widely recognized.

Therefore, it is needed to appropriately evaluate the risk of radioactive fission product release from the SFP. For the purpose of this, it is important to appropriately evaluate the fraction of fission product released from the SFP to the environment. This is done for a Severe Accident (SA) scenario where fuel damage has occurred in the SFP due to loss of cooling or a loss of SFP water. The SFP Level 3 PSA evaluates the risk to the public as a result of fission product release from SFP to the environment. To carry out a SFP Level 3 PSA, the amount of fission product released to environment as a result of a SA should be evaluated appropriately. In this study, it is evaluated using MAAP5 (Modular Accident Analysis Program, version 5) (Ref-1). In MAAP5, the fission product mass contained within each spent fuel bundle is assumed to be proportional to decay heat of that fuel bundle. The SFP contains cold fuel bundles which have been cooled in the SFP for a long time and hot fuel bundles which have been recently moved from the reactor core. As a result of assuming the decay heat of a fuel bundle is proportional to the fission products contained within it, the majority of short and long half-life nuclides will be contained within the hot fuel bundles. This assumption may lead overestimate of fission product release because a significant portion of the long half-life nuclides should be contained within the cold fuel bundles. Therefore, when a SA occurs in the SFP, where majority of the Cold fuel bundles maintain their integrity but the Hot fuel bundles fail, it results in most of the long half-life fission products being released to the environment and the release magnitude being overestimated.

As a result, this study proposes to use a revised method to modify the conservatism of fission product release in the MAAP5 code for the scenario where Cold and Hot fuel bundles are both present in the SFP. The fission product release is evaluated and examined using the proposed method.

II. REPRESENTATIVE SEVERE ACCIDENT SCENARIO

The SFP is located on the top floor of the reactor building in an ABWR (Ref-2), which is outside the primary containment. Therefore, if fuel damage occurs in the SFP, fission products are released into the reactor building. However, due to the large water pool volume in the SFP, there is a large time margin available to prevent fuel damage in the SFP (e.g. the time margin from the accident occurring to fuel damage is about 400 hours in the scenario where water is lost as a result of evaporation from the SFP due to the failure of heat removal and water injection systems). If appropriate accident management is conducted, most accident scenarios can be controlled without fuel damage.

Although there is small possibility of fuel damage in the SFP, it is important to consider mitigation of a SA in the SFP. The Plant Damage States (PDS) for the SFP is identified from the results of SFP Level 2 PSA.

II.A. Selection of Representative Severe Accident Scenarios

Plant Damage States (PDS) of the SFP PSA is defined corresponding to an initiating event. One is a Boil-off event which results in fuel damage due to a loss of SFP cooling following an initiating event. The second is a Loss of water inventory in

the SFP due to a leak. Drain of the SFP water is divided into two groups; Small(S) and Large(L) depending on the leakage area.

The occurrence frequency of the initiating event(s) that leads to these PDS (Loss of SFP Cooling and Loss of SFP inventory) are compared and examined (Ref-3). For Loss of SFP Cooling, malfunction of FPC system components such as a failure of the pumps and its supporting systems (RCW/RSW, air conditioning facility and power supply systems) are identified. The FPC and its supporting systems consist of many kinds of components that and many of them are active components such as pumps and fans. On the other hand, damage to the SFP Liner and Gate (including the damage from heavy load drop) are the reasons for Loss of inventory of pool water. Occurrence frequency of heavy load drop and the failure of passive components such as the Liner and Gate are small.

Therefore, the occurrence frequency for Loss of SFP cooling is much greater than that of Loss of inventory by comparing the occurrence frequency of the initiating events for both events. As a result, the Boil-off scenario becomes the dominant PDS in Level 2 PSA as well. For the reasons stated above, Boil-off case is selected as the representative scenario for this study.

III. SEVERE ACCIDENT ANALYSIS FOR SFP

This chapter describes the analysis and the results of the SA scenarios identified in chapter II.

III.A. Analysis Code

The analysis of the SFP in SA is conducted using MAAP5 (Ref-1). MAAP5 is a computer code that simulates the response of light water reactor (LWR) power plants including the SFP in SA. Given a set of initiating events and operator actions, MAAP5 predicts the reactor response as well as that of the SFP and the Reactor Building as the accident progresses. MAAP5 is an integrated code that treats a spectrum of important phenomena that could occur in SA, simultaneously modeling those that relate to the thermal-hydraulics and to the fission products. It also simultaneously models the primary system, containment, SFP, and the reactor/auxiliary building.

In the SFP model in MAAP5, spent fuel assemblies, spent fuel channels (racks), and the spent fuel pool are modelled.

- Spent fuel assemblies
  Each spent fuel assembly has a different decay power and mass. To simplify the problem, MAAP divides all the spent fuel assembles into a few power groups. Assemblies in a power group are assumed to be the same with regard to the decay power, masses of each material, and the geometry. The decay power in assemblies of a specific power group is calculated using decay power equations such as ANSI/ANS-5.1-1979 or REGULATORY GUIDE 3.54 (Ref-1).
- Spent fuel channels (racks)
  The spent fuel channel is similar to the spent fuel rack in concept. In the MAAP model, several channels can be modeled in one rack. The specific type of assembly and relevant information (e.g., burn up, time elapsed since discharge) are distributed in the channels and lumped together as a single heat source accordingly by "channel". In this sense, a "channel" as defined by MAAP represents a collection of spent fuel assemblies with varying characteristics all lumped together (by previously described "power groups") to form a single node.
- Spent fuel pool
  The spent fuel pool is modeled in a similar way as the reactor core. It is nodalised into channels in the cross-sectional direction and nodes in the axial direction.

MAAP5 contains models that calculate the release of fission products from the SFP. Additionally, it calculates the release of fission products to the reactor/auxiliary building and the environment through SFP leakage paths and failure paths. The fission product aerosol removal processes such as sedimentation, impaction and thermophoresis are considered in the MAAP code. It also calculates the decay energy generated by each group in each SFP rack.

III.B. Analysis Condition

TABLE I shows the SA analysis conditions in the SFP of an ABWR (Ref-2). Fig. 1 shows the nodalization of the SFP in the MAAP code. The SFP of the ABWR is located on the top floor (refueling floor) of the reactor building and a blowout panel is installed on the refueling floor. The blowout panel will be opened manually before a SA occurs in the SFP in order to dilute hydrogen gas concentration. The SFP contains fuels which have both a high decay heat power (Hot fuel bundle) and a low decay heat power (Cold fuel bundle). Hot fuel bundles are placed into the SFP shortly after the shutdown and they have a high decay heat power. Cold fuel bundles are cooled in the SFP for more than 18 months (1 cycle) and their decay heat power

is relatively low in comparison with the Hot fuel bundle. The decay heat power of the Hot fuel bundles is assumed conservatively to be the value of 4 days after the shutdown of the reactor. It corresponds to minimum time required to begin transferring fuel from the reactor to the SFP after reactor shutdown. In this case, it is assumed that hot bundles are to be loaded in the central racks of the SFP and the cold bundles are assumed to be loaded in the surrounding racks. In the Boil-off scenario, the time margin from the accident to fuel damage is about 400 hours. This is sufficient time to conduct SFP spray before fuel damage occurs. In this study, SFP spray is assumed to be conducted from 0.5 hour after fuel damage occurs.

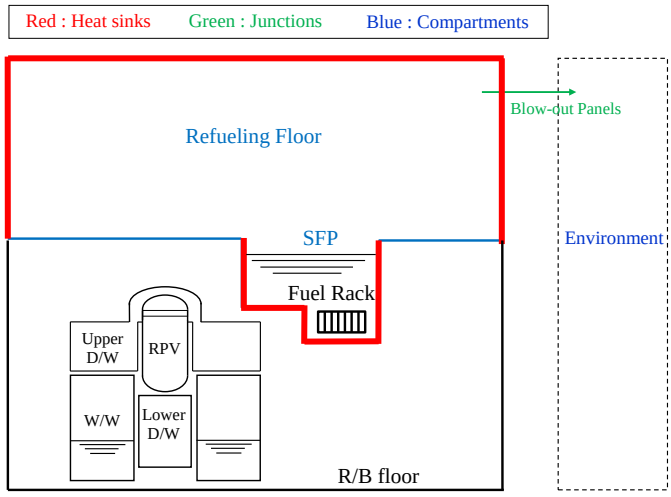


Fig. 1. Nodalization of SFP

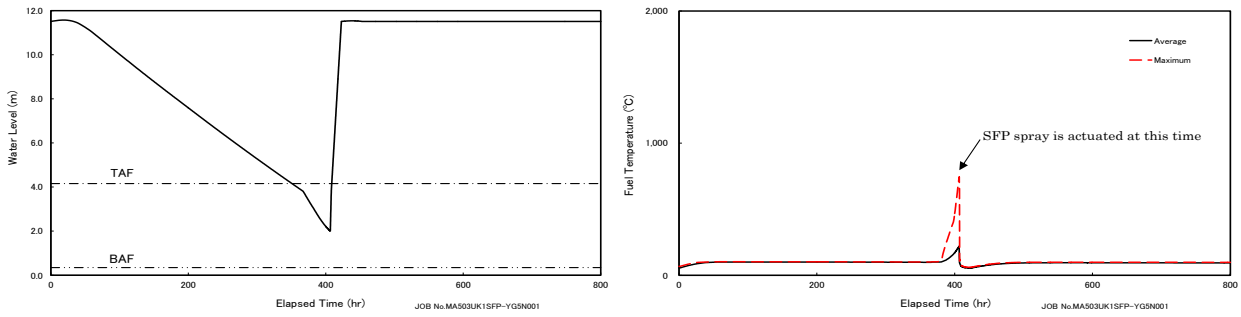
TABLE I. UKABWR Analysis Condition of SFP

Items	Conditions	Remarks
SFP Atmosphere	Air	-
Nodalization of SFP and R/B	SFP is located in the refueling floor connected to environment through blowout panel.	Fig. 1.
Initial Decay Heat	The decay heat is assumed to be value of 4 days after shutdown.	It corresponds to the decay heat power at the beginning of fuel transfer during shutdown.
Decay Heat Power	ANSI/ANS-5.1-1979 or REGULATORY GUIDE 3.54	-
Number of Fuel Nodes	Number of racks : 15 Axial Nodes : 27 nodes	11x11 bundles can be installed in a rack.
Initial Pressure	Atmospheric pressure	Environmental pressure is assumed.
Initial Gas Temperature	300 K	Environmental temperature is assumed.
Scenario	Boil-off	-
SFP Spray Status	Success	SFP spray is assumed to be conducted from 0.5 hours after fuel damage.

III.C. Analysis Result

This chapter shows the result of the analysis conducted using the conditions listed on TABLE I.

In this case, all heat removal and water injection systems are assumed to fail, however, SFP spray is assumed to be conducted from 0.5 hours after fuel damage. Time history of the water level is shown in Fig.2 (a). The water level goes down gradually due to boil-off of the water in the SFP. Fuel damage occurs at about 400 hours after the start of the accident. SFP spray is then conducted and the SFP water level is recovered. Fig.2 (b) shows the time history for both the mean and maximum value of the fuel temperature. Temperature of the Hot fuel bundle starts to rise as the water level decreases after it reaches TAF, which results fuel damage. Afterwards, when SFP spray injection is conducted, fuel temperature decreases rapidly. On the other hand, fuel damage does not occur to the Cold fuel bundles due to the decay heat power being relatively low. Fission products start to be released to the reactor building after fuel damage occurs, and is then released to the environment from the blowout panel. In this study, SFP spray is initiated after the hot fuel bundles are partially damaged but before any damage occurs to the Cold fuel bundles. As a result, only a part of the fission product is released to environment and the majority is retained within SFP rack.



(a) SFP Pool Water Level (b) Fuel Temperature in the Spent Fuel Pool

Fig. 2. Analysis Result

III.D. Discussion on the issue of Fission Product Release

As described in the introduction, MAAP assumes that fission product mass included in each fuel bundle is proportional to the decay heat for each fuel bundle. As this assumption might lead to an inappropriate result related to fission product release in some scenario, it was investigated whether the fission product release calculated from the MAAP code is appropriate or not by comparing the estimated fission product release.

In MAAP code, fission product release fraction to the environment is directly printed out as a parameter of FREL(n). The definition of FREL(n) is as follows:

$$FREL(n) = \frac{M_{rel}(n)}{M_{hot,t=0}(n) + M_{cold,t=0}(n)} \quad \text{(t : time [hr])} \tag{3-1}$$

Where,

$M_{rel}(n)$: Fission product mass released to the environment

$M_{hot,t=0}(n)$: Fission product mass included in hot bundles at the beginning of an accident

$M_{cold,t=0}(n)$: Fission product mass include in cold bundles at the beginning of an accident

n : Number of fission product group

On the other hand, the estimated fission product release fraction to the environment can be calculated using fission product mass reduction and initial inventory for hot bundles and cold bundles as follows:

$$F_{est}(n) = FM_{hot}(n) \times \Delta F_{hot}(n) + FM_{cold}(n) \times \Delta F_{cold}(n) \tag{3-2}$$

Where,

$FM_{hot}(n)$: Ratio of fission product mass included in hot bundle to total fission product mass in the SFP *

$FM_{cold}(n)$: Ratio of fission product mass included in cold bundle to total fission product mass in the SFP *

$\triangle F_m(n)$: Fission product mass reduction for hot bundles and cold bundles

$$\Delta F_m(n) = \frac{M_{m,t=0}(n) - M_{m,t=800}(n)}{M_{m,t=0}(n)} \quad \text{(m : hot or cold, t : time [hr])} \tag{3-3}$$

n : Number of fission product group

* : These values are calculated from initial inventory which is included in hot bundles and cold bundles. Initial inventory for each bundle can be calculated from the ORIGEN code. Initial inventory is the value where MAAP allocation is not applied.

As shown in Fig. 3, fission product release fraction, which is directly printed out as a parameter of FREL(n), is much larger than that the estimated fission product release fraction calculated from the formula (3-2). Therefore, if fission product release of the formula (3-1) is used as an input of Level 3 PSA, it might lead to conservative result for the risk evaluation of Level 3 PSA.

The following is the specific calculation method for the radioactivity released to environment using Level 3 PSA evaluation where the fission product release rate is obtained using the MAAP code, $FREL(n)$. As shown in formula (3-1), the fission product release rate obtained by MAAP, $FREL(n)$ is obtained as ratio of fission product mass released to environment at the end of analysis to the fission product mass which existed in the SFP the start of the analysis. As shown in TABLE II, the environmental release rate calculated using the MAAP code, $FREL(n)$ where the release rate of the entire fuel present in the SFP is calculated for fission product group, n, which is representative of each nuclide, i, present in that group. For example, $Bq_{rel}(i)$ [Bq], radioactive of nuclide, i, belongs to fission product group, n, released to environment, is :

$$Bq_{rel}(i) = FREL(n) \times (Bq_{hot,ini}(i) + Bq_{cold,ini}(i)) \tag{3-4}$$

Here, $Bq_{hot,ini}(i)$ is radioactivity of initial inventory for nuclide, i, contained in Hot fuel bundle and $Bq_{cold,ini}(i)$ is radioactivity of that in Cold fuel bundle. This evaluation method cannot evaluate the fission product release rate correctly because fission product mass is assumed to be allocated proportional to the decay heat of the fuel bundle.

From the reasons mentioned above, conservative result is expected if Level 3 PSA evaluation is conducted using fission product release rate FREL obtained by MAAP. Therefore, fission product release rate needs to be evaluated considering damage rate of the Hot and Cold fuel bundles.

TABLE II. The Relationship between MAAP Fission Product Groups and Radionuclides

MAAP FP Group	Representative Radionuclide	Radionuclide
Group 1	Nobles	KR 85
		KR 85M
		KR 87
		KR 88
		XE133
		XE135
Group 2	Iodine	I131
		I132
		I133
		I134
		I135
Group 3	Tellurium (dioxide)	TE127
		TE127M
		TE129
		TE129M
		TE131M
		TE132
Group 4	Strontium	SR 89
		SR 90
		SR 91
		SR 92
Group 5	Molybdenum	CO 58
		CO 60
		MO 99
		TC 99M
		RU103
		RU105
		RU106
		RH105
Group 6	Cesium	RB 86
		CS134
		CS136
		CS137
Group 7	Barium	BA139
		BA140
Group 8	Lanthanum	Y 90
		Y 91
		Y 92
		Y 93
		ZR 95
		ZR 97
		NB 95
		LA140
		LA141
		LA142
		PR143
		ND147
		AM241
		CM242
		CM244
Group 9	Cerium	CE141
		CE143
		CE144
		NP239
		PU238
		PU239
		PU240
		PU241
Group 10	Antimony	SB127
		SB129
Group 11	Tellurium	-
Group 12	Uranium (fuel)	-

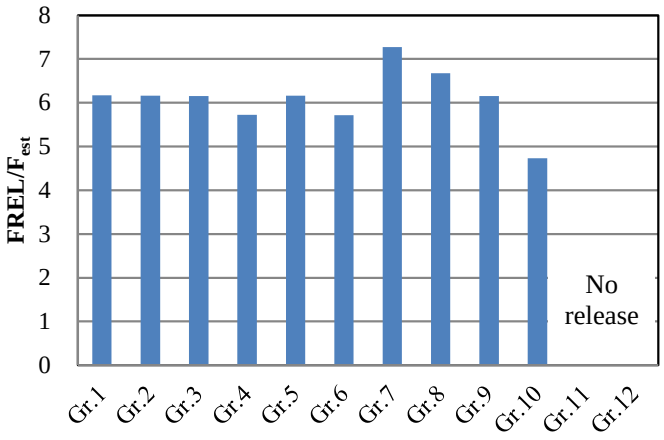


Fig. 3. Comparison between FREL and F_{est}

IV. PROPOSED METHOD FOR CORRECTING FISSION PRODUCT RELEASE

In light of the issues raised in the discussion above, this chapter will propose the practical evaluation method for the fission product release to environment using a formula for the fission product release rate for Hot and Cold fuel bundle instead of using Formula (3-4), which is generally used to calculate radioactivity for Level 3 PSA.

IV.A. Proposed Method

IV.A.1. Proposed Method for Cold Fuel Bundle

The method for the evaluation of fission product release to the environment for Cold fuel bundle is discussed. $F_{cold}(n)$ is the mass release rate from the Cold fuel bundle to the environment, and is obtained as follows:

$$F_{cold}(n) = \frac{M_{cold,ini}(n) - M_{cold,end}(n)}{M_{cold,ini}(n)} \quad (4\text{-}1)$$

Where $M_{cold,ini}(n)$ is the initial mass of fission product of group n in Cold fuel bundle and $M_{cold,end}(n)$ is the fission product mass of group n which is included in Cold fuel bundle at the end of the analysis.

Formula (4-1) assumes that all fission product released from Cold fuel bundle is released to environment (for example fission product in the melt pool is assumed to be released to environment). This conservatism in the fission product release to environment is done in case of a large damage to the Cold fuel bundle. However, it will give reasonable approximation for the cases where the damage is small.

From the reasons mentioned above, radioactivity of nuclide, i, released to the environment $Bq_{cold,rel}(i)$ [Bq] is:

$$Bq_{cold,rel}(i) = F_{cold}(n) \times Bq_{cold,ini}(i) \quad (4\text{-}2)$$

IV.A.2. Proposed Method for Hot Fuel Bundle

The method used in the evaluation of the fission product release to the environment from the Hot fuel bundle is provided below. The mass release rate $F_{hot}(n)$ from the Hot fuel bundle of fission product group n to the environment is shown as follows:

$$F_{hot}(n) = \frac{M_{rel}(n)}{M_{hot,ini}(n)} = FREL(n) \times \frac{M_{hot,ini}(n) + M_{cold,ini}(n)}{M_{hot,ini}(n)} \quad (4\text{-}3)$$

Where $M_{rel}(n)$ is the fission product mass of group, n, released to environment and $M_{hot,ini}(n)$ is the initial mass of fission product of group, n, which belongs to the Hot fuel bundle. Formula (4-3) has conservativism included by the fact that the Fission product mass includes small amount of the fission products released from Cold fuel bundle to the environment. However, it will give reasonable approximation in the case where the damage to the Cold fuel bundle is small. It is noted that release rate might be exceed 1.0 for the case of large damage of not only Hot fuel bundle but also major part of Cold fuel bundle, by the conservativeness of formula (4-3), however, the release rate is set as 1.0 for this kind of cases. The reason for the differences between the release rate formula (4-1) of Cold fuel bundle and that of Hot fuel bundle (4-3) are explained as follows. The decays heat of the cold fuel bundle is small as well as the degree of the fuel damage. Thus, in general, the formation of corium on the surface of SFP floor due to fuel melting and its subsequent release to the environment is unlikely to have occurred. On the other hand, the decay heat of the Hot fuel bundle is large and the fuel will melt if there is no mitigation, which results in a melt pool being formed on the surface of the SFP floor. Using formula (4-1), the nuclide in the corium pool formed outside of fuel is also included as the environmental release, and thus the result of evaluation will be exceedingly conservative if the same evaluation method of formula (4-1) is applied for Hot fuel bundle which forms corium. For these reasons, formula (4-3) is applied to obtain fission product release rate from Hot fuel bundle to the environment. Radioactive of nuclide i released to environment from Hot fuel bundle, $Bq_{hot,rel}(i)$ [Bq] is:

$$Bq_{hot,rel}(i) = F_{hot}(n) \times Bq_{hot,ini}(i) \quad (4\text{-}4)$$

IV.A.3. Summary of Proposed Method

TABEL III summarizes the general method and proposed method for calculation of radioactive release to environment.

TABLE III Summary of Proposed Method

	General Method (by FREL of MAAP to Level 3 PSA Evaluation)	Proposed Method (by the respective release from Hot and Cold to Level 3 PSA Evaluation)
Evaluation Method of Radioactivity released from SPF to Environment	$Bq_{rel,ori}(i) = FREL(n) \times (Bq_{hot,ini}(i) + Bq_{cold,ini}(i))$ Where, $FREL(n)$: Release rate to environment of MAAP fission product group n $Bq_{hot,ini}(i)$: Radioactive in the initial inventory of nuclide i, which included in Hot fuel bundle, obtained by ORIGEN $Bq_{cold,ini}(i)$: Radioactive in the initial inventory of nuclide i, which included in Cold fuel bundle, obtained by ORIGEN	$Bq_{rel,pro}(i) = F_{hot}(n) \times Bq_{hot,ini}(i) + F_{cold}(n) \times Bq_{cold,ini}(i)$ $= FREL(n) \times \frac{M_{hot,ini}(n) + M_{cold,ini}(n)}{M_{hot,ini}(n)} Bq_{hot,ini}(i) + \frac{M_{cold,ini}(n) - M_{cold,end}(n)}{M_{cold,ini}(n)} Bq_{cold,ini}(i)$ Where, $F_{hot}(n)$: Fission product Environmental release rate from Hot fuel bundle $F_{cold}(n)$: Fission product Environmental release rate from Cold fuel bundle $M_{hot,ini}(n)$: Fission product mass exists in Hot fuel bundle in early stage $M_{cold,ini}(n)$: Fission product mass exists in Cold fuel bundle in early stage $M_{cold,end}(n)$: Fission product mass exists in Cold fuel bundle at the analysis end n : MAAP fission product group n

IV.B. Result

TABLE IV shows the fraction of each nuclide present in the respective fuel bundles.

Fig.4 shows ratio of the result of fission product release evaluation by using the general and the proposed method. The values provided on the table are the releases calculated by using the proposed method which is then divided by the releases calculated using the general method. For short half-life nuclides in which hot inventory is sufficiently larger than cold one, the ratio of fission product release with this proposed method is more than 1.0 compared with the general method. The analysis result of this rationale is as follows. Evaluation formula for radioactive release of each nuclide with general method $Bq_{rel,ori}(i)$ [Bq] and that with the proposal method $Bq_{rel,pro}(i)$ [Bq] are:

$$Bq_{rel,ori} = FREL \times (Bq_{hot,ini} + Bq_{cold,ini}) \tag{4-5}$$

$$Bq_{rel,pro} = FREL \times \frac{M_{hot,ini} + M_{cold,ini}}{M_{hot,ini}} Bq_{hot,ini} + \frac{M_{cold,ini} - M_{cold,end}}{M_{cold,ini}} Bq_{cold,ini} \tag{4-6}$$

In this case, Cold fuel bundle is not damaged and obtained as $M_{cold,ini} - M_{cold,end} = 0$, thus the formula (4-6) is to be:

$$Bq_{rel,pro} = FREL \times \frac{M_{hot,ini} + M_{cold,ini}}{M_{hot,ini}} Bq_{hot,ini} \tag{4-7}$$

Therefore the ratio of radioactive release to environment with proposed and existing method is:

$$\frac{Bq_{rel,pro}}{Bq_{rel,ori}} = \frac{M_{hot,ini} + M_{cold,ini}}{M_{hot,ini}} \times \frac{Bq_{hot,ini}}{Bq_{hot,ini} + Bq_{cold,ini}} = \frac{Bq_{hot,ini}}{Bq_{hot,ini} + Bq_{cold,ini}} \Big/ \frac{M_{hot,ini}}{M_{hot,ini} + M_{cold,ini}} \tag{4-8}$$

As shown in Fig. 4, the result of formula (4-8) is more than 1.0 for short half-life nuclide and the reasons are as follows. Mass ratio of the Hot fuel bundle against whole mass as the denominator in formula (4-8) ($M_{hot,ini} / (M_{hot,ini} + M_{cold,ini})$) is about 0.8 in MAAP, with decay heat proportional distribution of fuel bundle as mentioned above. On the other hand, for radioactive ratio as numerator in formula (4-8) ($Bq_{hot,ini} / (Bq_{hot,ini} + Bq_{cold,ini})$) e.g. nuclide KR 85M is not existed in Cold fuel bundle and its radioactive ratio in Hot fuel bundle is 1 against whole fuel bundle. Therefore radioactive release evaluation using formula (4-8) is about 1.25. Decay heat proportional distribution of fuel bundle is for all nuclide in MAAP and about 20% is in Cold fuel bundle with this analysis condition. However, almost all short half-life nuclide is included in Hot fuel bundle and existing evaluation method lacks conservatism for the cases where there is small fuel damage in the Cold fuel bundles. Thus, by applying this proposed method, the under estimation of the fission product release is addressed. On the other hand, for long half-life nuclide which are contained within Cold fuel bundles more than in Hot fuel bundles (e.g. KR85, SR90), the ratio of fission product release with this proposal method is significantly reduced compared to the existing method. The reason can be explained as follows. The denominator in formula (4-8) ($M_{hot,ini} / (M_{hot,ini} + M_{cold,ini})$) is about 0.8 due to the assumption of fission product mass with decay heat proportional distribution of fuel bundles in the MAAP code. The fraction of inventory (Bq) of KR85 included in the Hot fuel bundle is about 0.2 and that of Cold fuel bundle is about 0.8. Therefore radioactive ratio as the numerator in formula (4-8) ($Bq_{hot,ini} / (Bq_{hot,ini} + Bq_{cold,ini})$) is about 0.2, and the radioactive release evaluation using formula (4-8) is about 0.2. Thus the effectiveness of this method is confirmed. It is noted that ratio of total radioactive release (Bq) with the existing method is approximately 0.8. The total radioactive release is reduced by about 20%. For the reasons mentioned above, in the case that there is no fuel damage to the Cold fuel bundles, fission product release from the SFP obtained by using MAAP is corrected by using this evaluation method and the release of short half-life nuclide being underestimated and that of long-life nuclide being overestimated can be evaluated respectively and appropriately.

TABLE IV Fraction of Inventory Included in Each Individual Bundle

MAAP FP Group	Representative Radionuclide	Radionuclide	Fraction of Inventory (Bq) Included in Each Bundles HOT (-)	COLD (-)
Group 1	Nobles	KR 85	0.2	0.8
		KR 85M	1.0	0.0
		KR 87	-	-
		KR 88	1.0	0.0
		XE133	1.0	0.0
		XE135	1.0	0.0
Group 2	Iodine	I131	1.0	0.0
		I132	1.0	0.0
		I133	1.0	0.0
		I134	-	-
		I135	1.0	0.0
Group 3	Tellurium (dioxide)	TE127	1.0	0.0
		TE127M	1.0	0.0
		TE129	1.0	0.0
		TE129M	1.0	0.0
		TE131M	1.0	0.0
		TE132	1.0	0.0
Group 4	Strontium	SR 89	1.0	0.0
		SR 90	0.2	0.8
		SR 91	1.0	0.0
		SR 92	1.0	0.0
Group 5	Molybdenum	CO 58	1.0	0.0
		CO 60	0.2	0.8
		MO 99	1.0	0.0
		TC 99M	1.0	0.0
		RU103	1.0	0.0
		RU105	1.0	0.0
		RU106	0.6	0.4
		RH105	1.0	0.0
Group 6	Cesium	RB 86	1.0	0.0
		CS134	0.4	0.6
		CS136	1.0	0.0
		CS137	0.2	0.8

MAAP FP Group	Representative Radionuclide	Radionuclide	Fraction of Inventory (Bq) Included in Each Bundles HOT (-)	COLD (-)
Group 7	Barium	BA139	1.0	0.0
		BA140	1.0	0.0
Group 8	Lanthanum	Y 90	0.2	0.8
		Y 91	1.0	0.0
		Y 92	1.0	0.0
		Y 93	1.0	0.0
		ZR 95	1.0	0.0
		ZR 97	1.0	0.0
		NB 95	1.0	0.0
		LA140	1.0	0.0
		LA141	1.0	0.0
		LA142	1.0	0.0
		PR143	1.0	0.0
		ND147	1.0	0.0
		AM241	0.0	1.0
		CM242	0.9	0.1
		CM244	0.1	0.9
Group 9	Cerium	CE141	1.0	0.0
		CE143	1.0	0.0
		CE144	0.7	0.3
		NP239	1.0	0.0
		PU238	0.1	0.9
		PU239	0.1	0.9
		PU240	0.1	0.9
		PU241	0.2	0.8
Group 10	Antimony	SB127	1.0	0.0
		SB129	1.0	0.0
Group 11	Tellurium	-	-	-
Group 12	Uranium (fuel)	-	-	-

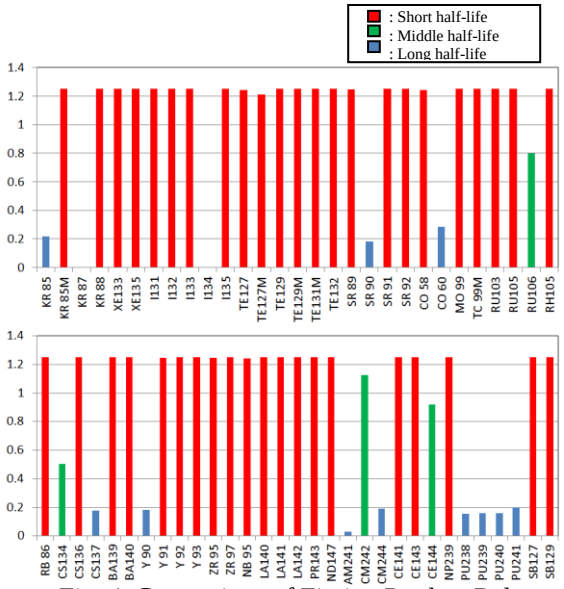


Fig. 4. Comparison of Fission Product Release by the General and the Proposed Method

V. SUMMARY AND CONCLUSION

In this study, a formula is proposed to modify the lack of conservatism for short half-life nuclides and the over conservatism for long half-life nuclides in the MAAP environmental release rate evaluation method. The following are the findings from this study:

- In the scenario where a SA occurs, which leads to the Hot fuel bundles in the SFP being damaged, but not the Cold fuel bundles due to the successful operation of the SFP spray beforehand, the following issues can be found in the existing MAAP evaluation method.
  1. In the MAAP code, fission product mass is assumed to be allocated proportional to the decay heat of the fuel bundle for all the nuclides. As a result, many of the long half-life nuclides that are supposed to be contained within the Cold fuel bundle are in the Hot fuel bundle. Thus, the environmental release of long half-life nuclides is overestimated as a result.
  2. On the other hand, a small amount of the short half-life nuclides which are supposed to be almost entirely contained within the Hot fuel bundle is contained within the Cold fuel bundle. The environmental release of short half-life nuclides is therefore underestimated.

- The following modifications are proposed as a countermeasure for the issues stated above:
  1. Modification formula of fission product release to environment from Cold fuel bundle is shown as (4-1)
  2. Modification formula of fission product release to environment from Hot fuel bundle is shown as (4-3)

- The following effects were confirmed as a result of applying the proposed method.
  1. Evaluation result of the radioactive release to the environment [Bq] in many of the long half-life nuclides are reduced more than 80% compared with the evaluation result obtained using the existing method.
  2. Evaluation result of the radioactive release to the environment [Bq] of the short half-life nuclides are increased by approximately 20% compared with the evaluation result obtained using the existing method.
  3. Total radioactive release [Bq] is reduced by approximately 20% compared with the evaluation result using the existing method.
  4. By applying the proposed method, the issue of the MAAP code allocating the fission products mass proportional to the decay heat of the fuel bundles for all nuclides is addressed. A more realistic Level 3 PSA evaluation will be available as a result.

- Further work can be carried in the future on this proposed method. For example, the following could be investigated;
  - For SA scenarios where even the cold fuel bundles are damaged and subsequently melt, fission product release rate can be overestimated because of the conservatism in the above proposed method. Therefore, development of a modification formula for the environmental release rate of fission products when using the MAAP code will be required for such scenarios.

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