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Consideration for Decision of Decay Heat Calculation Methodology on Spent Fuel during Loss of Spent Fuel Pool Cooling (LOSFPC)

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1. Introduction

IAEA's Specific Safety Requirements (SSR) publication has been revised by IAEA SSR-2/1 in 2016 [1]. This document has included the lesson-learn Fukushima Daiichi accident and the new concept "Multiple failure accidents" as shown in table I. The multiple failure accidents means postulated accident conditions that are not considered for DBA, but in the design process for the facility in accordance with the Best Estimate (BE) methodology, and in which release of radioactive material are kept within acceptable limits. In KOREA nuclear safety regulation, it is determined that the Loss of Spent Fuel Pool Cooling (LOSFPC) accident is included in the multiple failure accidents that must be considered as shown in table II.

Table I: Category of Accident Conditions

<table>
<tr><th colspan="2">Operational Status</th><th colspan="3">Accident Conditions</th></tr>
<tr><td rowspan="2">Normal Operation</td><td rowspan="2">Anticipated Operational Occurrences (AOOs)</td><td rowspan="2">Design Basis Accidents (DBAs)</td><td>Multiple failure accidents (Without core melt)</td><td rowspan="2">Severe accident (With core melt)</td></tr>
<tr><td>External hazards</td></tr>
<tr><td></td><td></td><td></td><td>Prevention of severe accident</td><td>Mitigation of severe accident</td></tr>
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Table II: The type of multiple failure accidents that must be considered

Classifications	Type of accident
Accidents that should be considered	AWTS
	SBO
	MSGTR
	TLOFW
	ISLOCA
	LOSCS
	LOUHS
	Loss of SI or recirculation with SBLOCA
	LOSFPC

In Nuclear Power Plants (NPPs), spent fuel assemblies discharged from the reactor core are generally stored in a Spent Fuel Pool (SFP) to ensure decay heat is low enough and prevent spent fuel heat-up. Therefore, the estimation of decay heat is essential to assure the integrity of SFP. In this paper, total decay heat and spent fuel uncovery time were estimated on various spent fuel conditions based on HANUL # 3 operation data and SFP cooling & cleanup system (SFPCCS) design methodology.

2. Key parameters for LOSFPC Analysis

In this section, the assumptions and initial conditions used to calculate decay heat are described. As mentioned above, the various conditions such as burnup, reactor operation period and down time should be considered for sensitivity analysis.

2.1 Reactor operation history for spent fuel assemblies grouping

Fuel assemblies are loaded in the core based on the loading pattern for getting a safe and economic core during the planned cycle length. According to this loading pattern, the spent fuel parameters such as burnup, enrichment are changed. Therefore, fuel group should be divided in accordance with loading pattern. In general, the fuel loading pattern for PWR is low-leakage 3-batch loading and the fuel group can be divided into three group on full core as shown in figure 1.

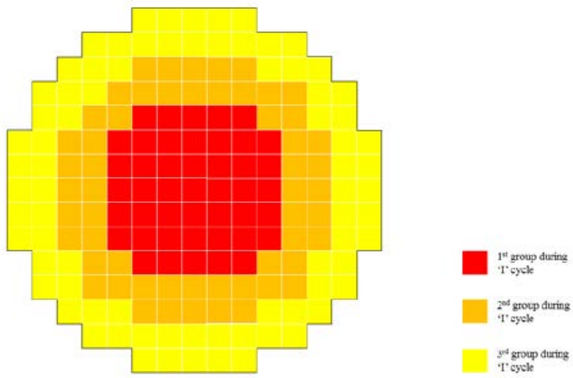

Fig. 1. Typical loading pattern for non-leakage 3-batch loading

According to this loading pattern design, spent fuel assemblies are divided into three (3) groups on each cycle and the spent fuel assemblies grouping can be divided as shown in figure 2 and table III. Actually, each of the spent fuel assemblies has a little different enrichment. However, it is negligible and it is assumed that these groups have almost same burnup, enrichment and operation period.

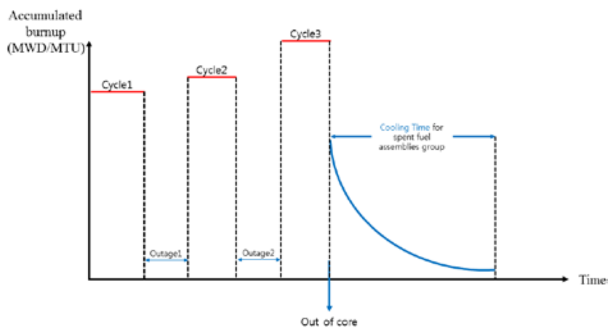

Fig. 2. Typical operation and down time cycle for spent fuel

Table III: The group of spent fuel according to operating cycle

Batch \ Cycle	1	2	3	…
1st	C	D	E	…
2nd	B	C	D	…
3rd	A	B	C	…

The assumption of operation mode at initiation of accident is also important to estimate the decay heat. Generally, it is considered that three (3) operation modes are normal, abnormal and refueling modes to estimate the spent fuel pool accident. In this paper, only refueling mode was considered.

2.2 Calculation standards for decay heat calculation

With the LOSFPC, the water heats up and begins to boil, which ultimately leads to a reduction in the water level and inventory. In this situation, the decay heat is dominant factor. For many years, the standards for calculating decay heat have been developed such as Auxiliary Systems Branch (ASB)-9.2, Regulatory Guide 3.54, ANSI/ANS-5.1 and so on.

2.2.1 Auxiliary Systems Branch (ASB)-9.2

The U.S. NRC developed this calculation method based on ANSI/ANS-5.1-1971 to evaluate the spent fuel pool cooling. This calculation method is not just simpler than the ANSI/ANS-5.1-1971 but also more conservative. It was applied to the Standard Review Plan (SRP) and Korea nuclear safety regulation document [2, 3]. However, when this calculation were announced, it had typographical error in equation 2 so that the equation was corrected in equation 3 in recent years [4].

$$P/P_0\,(\infty, t_s) = \frac{1}{200}\sum_{n=1}^{11}[A_n e^{-a_n t_s}] \quad [1]$$

$$P/P_0\,(t_0, t_s) = (1+K)\frac{P}{P_0}(\infty, t_s) - \frac{P}{P_0}(\infty, t_0+t_s) \quad [2]$$

$$P/P_0\,(t_0, t_s) = (1+K)(\frac{P}{P_0}(\infty, t_s) - \frac{P}{P_0}(\infty, t_0+t_s)) \quad [3]$$

2.2.2 Regulatory guide 3.54

The U.S. NRC developed this calculation standard for the storage of spent fuel in an Independent Spent Fuel Storage Installation (ISFSI). This regulatory guide applied the ORIGEN-S code and interpolation of ANSI/ANS-5.1 table data. However, the revision draft was published in 2016 and calculation method has been changed [5]. The revised calculation method applies the combination of ISO-10645 and ANSI-ANS-5.1 methods considering the decay heat fraction generated on fission products.

2.2.3 ANSI/ANS-5.1

The ANSI/ANS-5.1 was enacted in 1971 and applied to meet the conservative evaluation for Emergency Core Cooling System (ECCS) in 10 CFR 50.46. The revised version was announced in 1973, 1994, 2005 and 2014.

3. Geometrical modeling for MAAP5

The SFP geometrical model has been developed using the design data for the spent fuel and fuel handling building of the HANUL #3. Variables for geometry of pool, rack and fuel handling building are referred to the design documents. The spent fuel storage racks are stainless steel structures of rectangular cells which have poison sheets are attached on the outside. The rack modules use a two region modular design composed of Region I and II. In case of HANUL #3, the SFP has 16 racks and about 1500 cells [6]. Radial nodalization in the MAAP SFP refer to the radial arrangement of the SFP racks relative to one another and the SFP structural walls. Once the racks begin to have significant interaction with adjacent and or the SFP walls it is necessary to have a detailed radial model in order to account for the temperature gradients which can develop. Therefore, the channels are subdivided into 36 channels in as shown in figure 3.

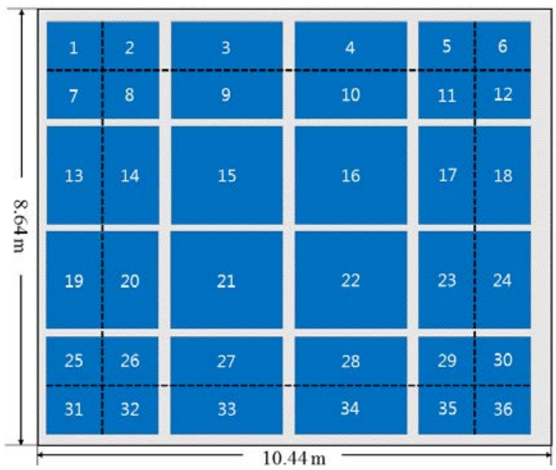

Fig. 3. Plan of the spent fuel racks model for modeling

The axial direction nodes of spent fuel assembly were divided into 32 nodes. As shown in figure 4, fuel assembly has non-fuel region and active core region.

Therefore, the top and bottom axial nodes simulate the lower non-fuel region and nodes 2 to 31 denote the active fuel region.

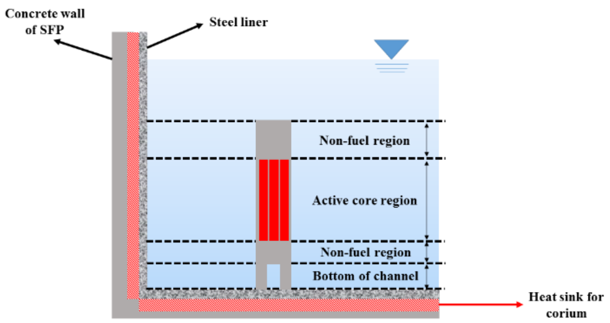

Fig. 4. Axial diagram of spent fuel assembly for modeling

The fuel handling area modeling is an important feature in the MAAP SFP model because the geometry input for the SFP pit will dictate the initial water mass in the SFP. The initial water mass in the SFP pit is a crucial component because the large volume of water contains the thermal inertia which must be heated in order for the water to begin boiling and eventually steaming away to cause a LOSFPC. The fuel handling area is not air tight and typically contains many leakage paths and or designed failure panels to prevent over pressurization of the auxiliary building. Therefore, a junction from the fuel handling area should be modeled as shown in figure 5 to ensure that the pressure in the MAAP model remains physical for a LOSFPC in the SFP.

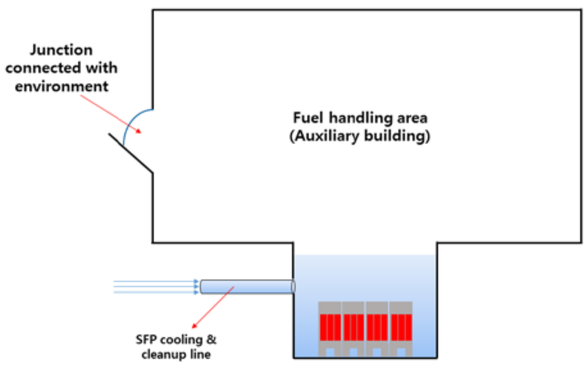

Fig. 5. Schematic diagram of fuel handling building for modeling

4. Sensitivity Analysis and Results

4.1 Operation history assumptions for Sensitivity analysis

As mentioned above, the sensitivity analysis of operating history was performed to estimate decay heat and spent fuel uncovery time on various spent fuel conditions. The sensitivity cases are described as follows:

- Case I: Thermal power + uncertainty 2%,
  Down time 0.0 s

- Case II: Thermal power + uncertainty 2%,
  Down time = operation time - EFPD,
  EFPD (Effective Full Power Day) = 476 days

- Case III: Thermal power, HANUL #3 operation data (Transient core design + equilibrium core design)
  Down time = 31 days

As shown in table IV, the core design parameters are same to simulate the equilibrium core in case of I and II. These assumptions are considered based on the design data for cooling capacity of SFPCCS. It is more conservative than realistic operation data.

Table IV: Common assumptions for case I and II [7]

Parameters	Values
Thermal power (MW_t)	2,815
Operation time	18 months
full core on last cycle discharged time	100 hrs after shutdown
Accumulated burnup (GWD/MTU)	18(cycle1), 36(cycle2), 54(cycle 3 in figure 2)
Enrichment (%)	4.4

Table V: Assumptions the equilibrium core for case III

Parameters	Values
Thermal power (MW_t)	2,815
Operation time	1.5 years(=547.5 days)
full core on last cycle discharged time	100 hrs after shutdown
Accumulated burnup (GWD/MTU)	21(cycle1), 40.5(cycle2), 50(cycle 3 in figure 2)
Enrichment (%)	4.5

4.2 The assumptions of Reactor operation mode and spent fuel assemblies inventory

As mentioned in chapter 2.1, the refueling mode was selected to estimate the decay heat conservatively. In case of refueling mode, it is assumed that full core is discharged at the end of last cycle. As shown in table VI, 68 spent fuel assemblies was discharged from cycles 1 to 19 and 177 spent fuel assemblies was discharged on last 20th cycle. The number of cycles is considered based on the SFP storage volume for HANUL #3. However, accumulated burnup of discharged fuel assemblies on last cycle is different. Therefore, it is considered that the discharged fuel assemblies on last cycle is divided into three groups.

Table VI: Discharged fuel assemblies in core of each cycle (Case I and II)

Group	Cycle						
	1	2	3	…	20		
A(1st)	68			…			
B(2nd)		68					
C(3rd)			68				
…							
T(20th)					59		

U(21st)						59	
V(22nd)							59

In case of case III, it was applied that the spent fuel pool was storing 607 spent fuel assemblies for 13 cycles based on HANUL #3 operation and refueling data. Additionally, the 1/3 core from 14 to 25 cycles were discharged and the full core on last cycle was discharged. Therefore, there were 607, 59 X 12 and 177 assemblies being stored in spent fuel pool. The number of cycles is also considered based on the SFP storage volume for HANUL #3.

Table VII: Discharged fuel assemblies in core of each cycle (Case III)

Group	Enrichment (%)	Cycle							
		1	2	3	4	…	26		
A(1st)	2.4	45				…			
B(2nd)	3.0		53						
C(3rd)	3.5			64					
D(4th)	4.0				48				
…	…								
N(26th)	4.5						59		
O(27th)								59	
P(28th)									59

4.3 Sensitivity analysis results for operation history

As mentioned in chapter 3.1 and 3.2, the operation history was assumed in accordance with conservative methodology and realistic operation data. The sensitivity result is shown in figure 6 and table VIII. Total decay heat in case II is larger than other cases. The case III which is considered based on realistic data have lowest decay heat. The conservative methodology considered more spent fuel assemblies and assumed equilibrium cycle. On the other hand, it was applied that the realistic data has initial transient core and irregular down time.

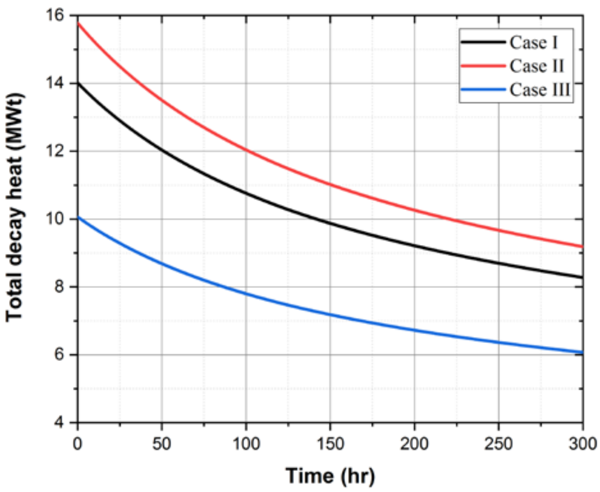

Fig. 6. Total decay heat of spent fuel for each case

Table VIII: Total decay heat at 100 hours after shutdown

Case	Decay heat (MW_t)
I	14.0
II	15.7
III	10.1

The collapsed and mixture level in SFP is presented in figure 7. The collapsed water level increases at the beginning of the LOSFPC sequence because of the expansion of pool water with increasing temperature. In case I, top of the spent fuel assemblies were exposed after 43.5 hours and the active core region uncovered after 45 hours. In case II, top of the spent fuel assemblies were exposed after 38.4 hours and the active core region uncovered after 40 hours. And in case III, also the uncovery of spent fuel assemblies and active core region occurred after 62 and 64.5 hours, respectively.

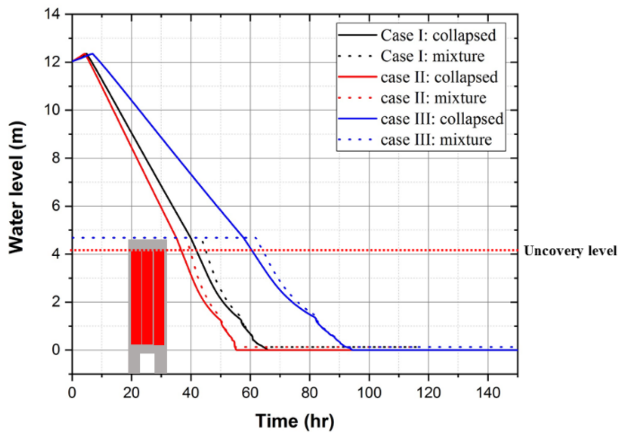

Fig. 7. Collapsed and mixture water level in SFP for each case

4.4 Sensitivity analysis results for decay heat calculation standard

As mentioned in chapter 2.1, various decay heat calculation standard was developed and most of thermal-hydraulic system code and severe accident code have the ANSI/ANS-5.1 calculation options. In this paper, the ANSI/ANS-5.1 sensitivity calculation was performed using the options of MAAP5. The assumptions of case III from chapter 4 was used for sensitivity analysis. To calculate the ANSI/ANS option using MAAP5, the power fractions for fission of U-235, Pu-239, U-238 and Pu-241 should be changed according to the burnup and enrichment. Therefore, the power fractions referred to the 3rd revision draft document of regulatory guide 3.54 [5]. Additionally, the ANSI/ANS-5.1 (2014) which is the newest version of ANSI/ANS-5.1 is compared with calculation result of other ANS calculation and SCALE6.1/ORIGEN-S.

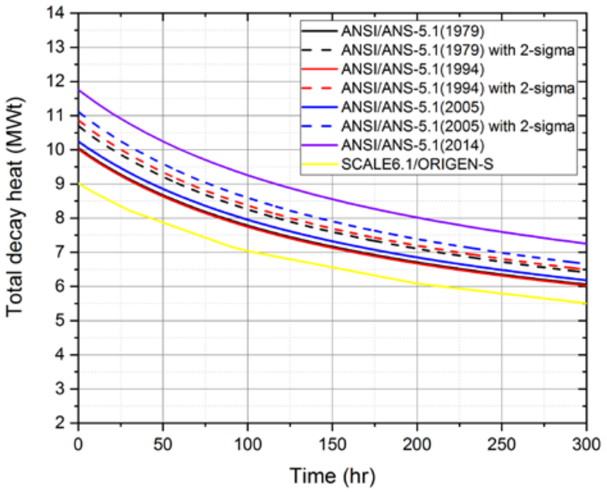

Fig. 8. Comparison of total decay heat with ANSI/ANS standards

As shown in figure 8, the most conservative result is about 11.7 MW_t in case of ANSI-ANS-5.1 (2014) case. The ANSI-ANS-5.1 (1994) without 2-sigma case is about 10 MW_t which is lower than other ANSI/ANS calculation results. The result of ORIGEN-S calculation was about 9 MW_t which is 1 MW_t lower than the ANSI-ANS-5.1 (1994) without 2-sigma case.

5. Summary and Conclusions

In this paper, various conditions regarding the LOSFPC were studied by using the MAAP5 code. According to the analysis results, the methodology which is considered at designing the cooling capacity of SFPCCS have more conservative result. In sensitivity case of decay heat calculation standard, the ANSI-ANS-5.1 (2014) case has the most conservative result and the result of ORIGEN-S calculation is lower than ANS standard results. Based on the results of sensitivity analysis in this paper, it is confirmed that the assumptions of operation history and spent fuel assembly inventory caused the different time margin to take action during the LOSFPC. These results will be considered to determine the BE methodology of decay heat calculation.

ACKNOWLEDGMENTS

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