Dose Assessment for OPR1000 during Loss of Spent-fuel-pool cooling

KYUNGHO NAM[1*], Seung Chan LEE[2]

[1,2]*KHNP CRI, 70, Yuseong-daero 1312 beon-gil, Yuseong-gu, Daejeon, Korea, 34101*
*Corresponding author's e-mail: khnpkhnam@khnp.co.kr

Abstract In accidents known as "loss of spent-fuel pool cooling," the cooling system does not operate and the water level decreases due to the decay heat released from the stored spent-fuel assemblies in the spent-fuel pool. In this paper, a dose assessment for the evaporated radioactive material was performed while the water level reached the top of the fuel assemblies in the OPR1000 (that is, the optimized power reactor) in Korea. First, an assessment of the time taken to reach the boiling temperature and water level corresponding to the top of the fuel assemblies was performed. Next, the maximum evaporation rate was calculated. Based on these calculated results, a dose assessment was performed. For a conservative assessment, the minimum time to reach the water level corresponding to the top of fuel assemblies was applied. Additionally, it was also assumed that the mitigation strategy (injection of water into the spent-fuel pool) was not implemented. The specific activity data about the coolant in the spent-fuel pool was referred to the specific design document for the OPR1000. As a result, the effective dose met the required acceptance criteria during the loss of spent-fuel-pool cooling accident.

KEYWORDS: Spent-fuel Pool, Loss-of-cooling, Effective Dose.

1 INTRODUCTION

Following the Fukushima nuclear accident in 2011, there were many changes to safety design criteria and/or regulations around the world based on the lessons learned from the accident. The International Atomic Energy Agency (IAEA) Specific Safety Requirements (SSR) notes that the design basis of safety features for Design Extension Conditions (DEC) is beyond that of a Design Basis Accident [1]. The Nuclear Safety and Security Commission (NSSC) in Korea has required plant-specific accident management plans, which are extended to a BDBA, including severe accidents. According to law revised by the NSSC in Korea, for the multiple failure accidents involved in DECs, the effective dose should not exceed 250 mSv.

One of the important subjects of the Design Extension Condition (DEC) accident evaluation is Spent-fuel Pool (SFP) safety due to loss of the cooling function after multiple failures. The Fukushima nuclear accident highlighted the need for more attention to the SFP. The ability to cool the spent-fuel assemblies in the SFP and maintain adequate temperatures is a critical aspect of preserving safe conditions both on site and off site. The general purpose of the fuel-pool cooling system is as follows. First, the fuel-pool cooling system maintains the water level in the fuel pool alone or in all pools during normal and reloading operation. Second, the fuel-pool cooling system provides water of high clarity and purity to the pools and removes any radioactive contamination released to the water during normal handling of irradiated fuel and reactor vessel components. Last, this system removes decay heat from irradiated fuel assemblies to maintain the pool water temperature below approximately 60 °C under normal conditions. Radiation shielding using the SFP inventory is also needed for habitable locations adjacent to the SFP. The loss of SFP cooling results in overheating and ultimately reaching or approaching saturation temperatures. Finally, the water inventory is reduced and the fuel assemblies uncovered if no mitigation strategies are applied. For these reasons, the multiple-barrier accident-coping strategies were established and the external injection strategy for loss of SFP cooling accidents was selected in Korea.

The purpose of the work reported in this paper was to evaluate the effective dose that resulted from the loss of SFP cooling of the OPR1000-type nuclear power plant in Korea. The period for evaluating the effective dose was the time needed to reach the water level corresponding to the top of the spent-fuel assembles from the time needed to reach the water temperature corresponding to boiling. The dose evaluation in the case of fuel uncover (which is of concern during severe accidents) is not considered in this paper.

2 ININITIAL CONDITIONS FOR DOSE ASSESSMENT

2.1 Assumptions for the accident scenarios

In this work, the selected scenarios related to the number of stored spent-fuel assemblies were the normal, re-loading, and abnormal scenarios following the operation mode.

In the first (normal) scenario, an accident occurs during a period of normal operation. When the normal operation is ongoing, only previously discharged spent-fuel assemblies are stored in the SFP and the storage capacity for a full core of fuel assemblies is retained for re-load operation. Therefore, the number of previous operation-cycles and discharged fuel assemblies was determined considering the design capacity of the SFP, except for the number of full-core fuel assemblies. The number of discharged fuel assemblies per cycle corresponds to one-third of the core capacity. Considering the design capacity of the SFP rack, it was determined that there were 18 previous operation cycles. This assumption applies equally to the other scenarios.

In the second (re-loading) scenario, the assumption was determined following the plant-specific design document [2]. According to the design document, the SFP cooling trains for the spent-fuel pool must have a heat-removal capacity corresponding to the total decay heat released by freshly discharged fuel assemblies (during the one-cycle re-load operation) and by previously stored fuel assemblies. Moreover, these trains also have heat-removal capacity corresponding to 100 % of the designed decay-heat load so that the water temperature in the SFP remains below 60 °C. The stored fuel assemblies in this scenario are previously discharged fuel assemblies corresponding to one-third of the core (discharged in each cycle) and freshly discharged fuel assemblies corresponding to the full core discharged for the re-load operation.

In the third (abnormal) scenario, an operation mode is postulated for a conservative evaluation. This scenario includes the re-load scenario with additional fuel assemblies corresponding to one-third of the core capacity. It was assumed that a reactor trip suddenly occurs and that fuel assemblies corresponding to one-third of the core are discharged and stored in the SFP. Reactor operation begins after the spent-fuel transportation. However, the reactor trip re-occurs and fuel assemblies corresponding to full-core are discharged and also stored in the SFP. Because of these conservative assumptions, this scenario covers both the normal and re-loading scenarios, as mentioned above.

The maximum decay heat was determined according to the mentioned assumptions, and the result is described in Table 1. Generally, the decay heat would gradually decrease over time, but it is conservatively assumed that the decay heat is constant in this paper.

Table 1: Loss of SFP cooling accident scenario for evaluating the effective dose

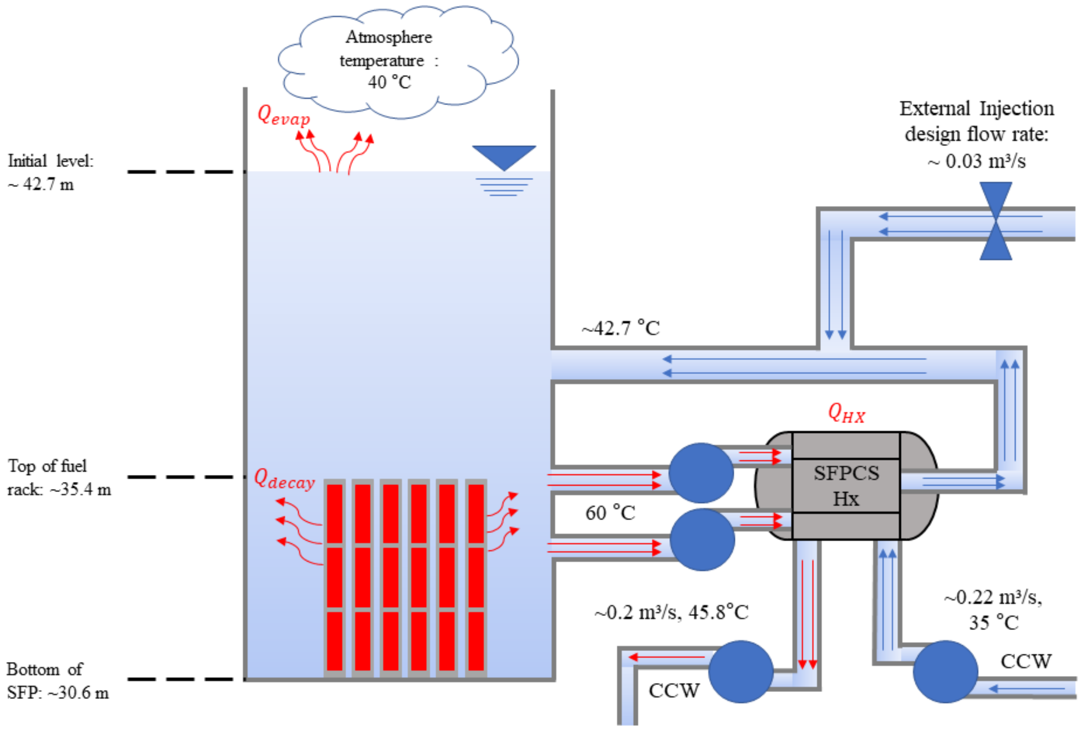
Mode	Accident scenarios to evaluate effective dose		
	Number of stored fuel assemblies	Maximum decay heat (MWt)	Number of operating cooling train
Normal	1,292	5.6	1
Re-loading	1,401	11.1	1
Abnormal	1,469	13.2	2

2.2 Calculation of the initial water temperature in the SFP

This evaluation was performed to determine the water bulk temperature in the SFP during the postulated discharge of spent-fuel assemblies, as a function of time. The mathematical formulation for this evaluation can be explained with reference to the simplified heat-exchanger alignment shown in Figure 1. Referring to the SFP cooling system, the governing differential equation can be written by utilizing the conservation of energy as described in equation 1.

$$C_{SFP} \times m_{SFP} \times \frac{dT}{dt} = Q_{decay} - Q_{HX} - Q_{evap} - Q_{cond} - Q_{piping} \quad (1)$$

Figure 1: The schematic diagram of SFPCS and major variables related to evaluation



Heat removal from the SFP is accomplished by forced circulation of cooling water in the SFP, heat loss from the SFP cooling system piping, natural convection and evaporative cooling from the poll surface, and conduction through the concrete walls of the pool. Where, C_{SFP} in equation (1) is the heat capacity of the water in SFP, m_{SFP} is the water mass in SFP, Q_{decay} is the total decay heat released by the stored spent-fuel assemblies as described in Table 1, Q_{HX} is the rate of heat removal by the heat exchanger of SFP cooling system, and Q_{evap} is heat loss by evaporation from the pool surface. Q_{cond} is the heat loss by conduction through the pool wall and Q_{piping} is the heat loss through the SFP cooling system piping. However, the Q_{cond} and Q_{piping} are very small and without major effect, so it was conservatively neglected. The heat transfer rate through the heat exchanger can be calculated using equation (2).

$$Q_{HX} = W_{HX} \times C_{HX} \times \frac{T_{HX,o} - T_{HX,i}}{T - T_{HX,i}} \times (T - T_{HX}) \quad (2)$$

Where, W_{HX} is the mass flow rate at heat exchanger inlet, C_{HX} is the heat capacity of coolant in the heat exchanger, T is the pool (water) bulk temperature, T_{HX} is the coolant temperature in the heat exchanger. The expression ($T_{HX,o} - T_{HX,i}$ / $T - T_{HX,i}$) was used to determine the temperature effectiveness of the heat exchanger. In the abnormal scenario, the heat removal rate of the heat exchanger doubles if the assumption mentioned in Table 1 is followed. The evaporative heat loss, Q_{evap} is a nonlinear function of the water temperature in the SFP, and is evaluated at the maximum fuel-handling-building room temperature. The mass evaporation rate, $M(T,T_A)$ is needed to calculate the evaporative heat loss and the equations utilized are as follows.

$$Q_{evap} = M(T,T_A) \times h_{fg} \times A_S + h_c \times (T - T_A) \times A_S \quad (3a)$$

$$M(T,T_A) = -64.8 \times \left(\frac{C(T_A) \times D_{H_2O,air}(T)}{L}\right) ln\left(1 - (P_{sat}/P_{atm})\right) \quad (3b)$$

$$h_c = 1.32 \left(\frac{T - T_A}{L_c}\right)^{1/4} \quad (3c)$$

Where, h_{fg} is the latent heat of the pool water, A_s is the pool surface area, T_A is the ambient air temperature, $C(T_A)$ is the mol number per volume, $D(T)$ is the diffusion coefficient, P_{sat} is the saturated pressure, P_{atm} is the atmospheric pressure, and h_c is the convection-heat transfer coefficient for laminar flow at the pool surface. The initial bulk temperature of the pool was calculated using the formulas mentioned above, and the calculation results are presented in the following table.

Table 2: Calculated initial water temperature at initiation of loss of SFP cooling

	Accident scenarios to evaluate effective dose		
	Normal	Re-loading	Abnormal
Initial temperature(°C)	45.6	58.4	49.53

2.3 Evaluation of major sequence time and evaporation rate

2.3.1 Calculation of major event time during loss of SFP cooling

Estimates of the sequence time to boiling and time to fuel uncovering are a function of the initial pool temperature, pool volume, and the makeup of the fuel used in the pool. It was conservatively assumed that no makeup water strategy was available during this period. First, the time to reach the boiling point was calculated according to the following equation.

$$Q_{decay} \times t_{boil} = V_{SFP} \times \rho_{SFP} \times C_{SFP} \qquad (4)$$

Where, t_{boil} is the time needed to reach the boiling point (100 °C), V_{SFP} is the net volume of the water in the SFP, and ρ_{SFP} is the density of the SFP water. The net volume of water was calculated using the specific design document for the SFP rack, and this volume was about 960 m^3. This value was calculated by excluding the volume of the fuel-storage rack and stored spent-fuel assemblies. The water density was about 983 kg/m^3, corresponding to 60 °C (the maximum temperature according to the design requirement).

$$t_{rack} = t_{boil} + \frac{h_{fg} \times V_{rack} \times \rho_{SFP}}{Q_{decay}} \qquad (5)$$

Where, t_{rack} is the time needed to reach the water level corresponding to the top of the spent-fuel assemblies and V_{rack} is the water volume of the upper fuel storage rack (~ 700 m^3).

2.3.2 Evaporation rate

Last, the evaporation rate was calculated using equation (6). Where, V_{max} is the maximum evaporation rate.

$$\dot{V}_{max} = \frac{V_{rack}}{t_{rack} - t_{boil}} \qquad (6)$$

The initial condition was calculated as mentioned in Section 2, and the calculated results are given in Table 3.

Table 3: Calculated initial conditions for evaluating the effective dose

	Variables for initial condition		
Mode	Time to reach the 100°C of boiling point (hours)	Time to reach the water level corresponding to top of spent-fuel assemblies (hours)	Maximum evaporation rate (m^3/hr)

Normal	10.63	87.4	9.1
Re-loading	4.11	42.98	18.0
Abnormal	4.18	36.73	21.5

3 DOSE ASSESSMENT AND RESULTS

During loss from the SFP water inventory, there is considerable potential to reach the water level of the top of the spent-fuel assemblies. In this study, three scenarios were considered for dose estimation, as in the previous section.

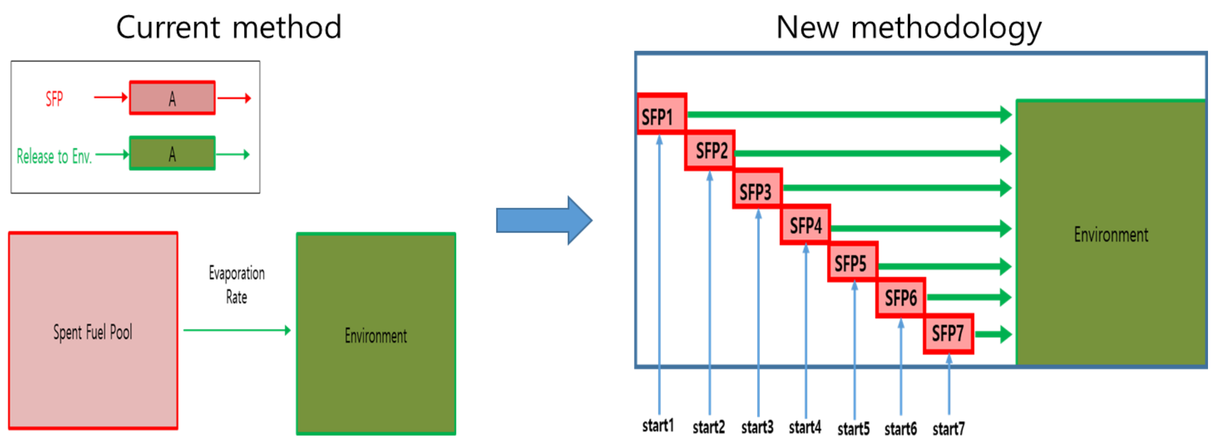
3.1 New methodology of the radiological analysis

In this study, a new methodology used for dose estimation involved RADTRAD 3.03. The results were compared with hand calculated results to confirm the conservatism of the results. RADTRAD 3.03 includes various functions such as isotope-decay phenomena, time delay, and flashing behavior, but this code commonly uses the constant compartment-volume model. This model generally has some dilution effect in the case of a decrease of the SFP water inventory. In this study, to overcome the non-conservatism of this model, a new modeling method was used, named the infinite compartment-volume model. This method was first calculated using RADTRAD.

3.2 Estimation concept

For the current radiological estimation, the SFP was conservatively hand-calculated. Here, the hand calculation and RADTRAD model results are compared using the new modeling method. The dilution effect of RADTRAD was removed by the multiple volume methods, which are illustrated below.

Figure 2: The multiple volume estimation of SFPCS by RADTRAD new methodology



As shown Figure 2, the new RADTRAD methodology has multiple evaporation volumes and the SFP water in each volume is evaporated step-by-step (from SFP1 to SFP7). Each evaporation start time is labeled (i.e., start1, start2, start3, start4, start5, start6, start7).

3.3 Verification of the new methodology and dose estimation results

From Section 3.2, we calculated the radiological effects using the new RADTRAD methodology. In these calculations, seven starting points were used to calculate the SFPCS radiological effects. This means that the total volume of the SFP was divided into seven volumes. Each volume has a different evaporation release start-time (which proceeded step-by-step). The release time of SFP2 is started at “start2” after the SFP1 starting time “start1”. As each step ended, the next starting time (e.g., “start3”) is started. As shown in Figure 2, the last step (“start7”) is the final calculation process. Table 4 shows that the new RADTRAD methodology results in the three scenarios of the SFPCS radiological effects. In addition, Table 5 and Figure 3 show a comparison of the test results for hand calculation and the RADTRAD models.

From these results, the new RADTRAD methodology was verified by achieving good agreement with the hand-calculation results. In this study, the new RADTRAD methodology strongly overcame the dilution effects of the current RADTRAD model and showed good agreement with the hand calculation. As shown in Table 5, the RADTRAD model (new methodology) result is more conservative than the hand-calculation result and compared values matched within the 0.2%.

Table 4: Calculated results of 3 scenarios in SFP inventory decrease from Table 3

Scenario Case	TEDE(Total Effective Dose Equivalent) : mSv		
	External	Internal	Total
Normal	6.26e-04	6.59e-02	6.65e-02
Re-loading	1.24e-03	1.30e-01	1.32e-01
Abnormal	1.48e-03	1.56e-01	1.57e-01

Figure 3: The comparison of each methodology (Hand calculation vs RADTRAD)

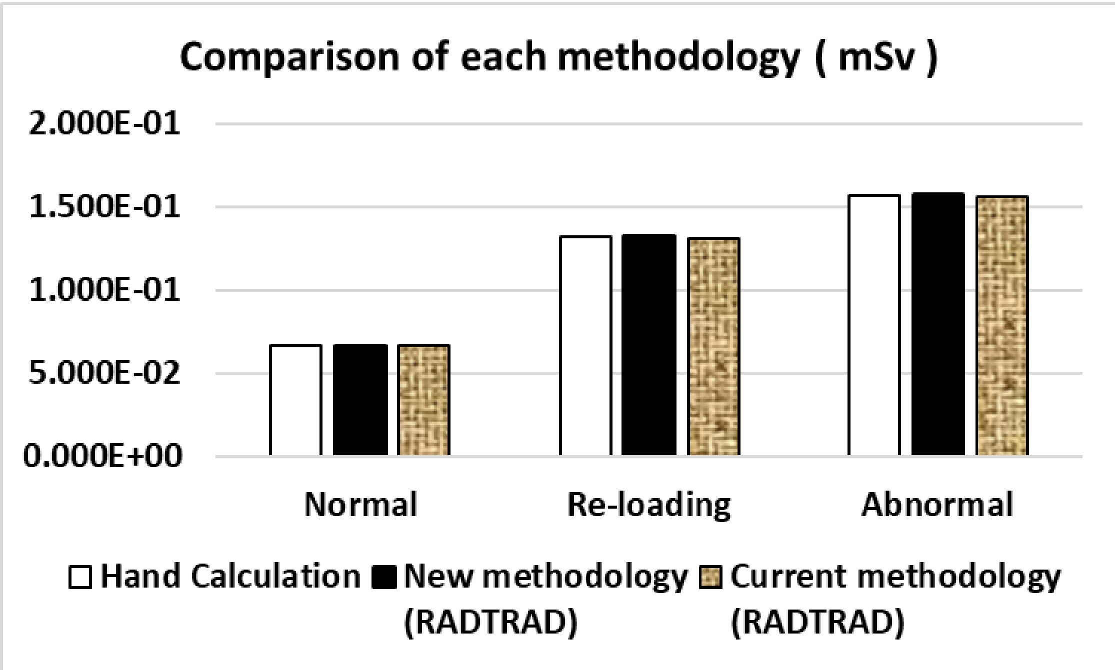


Table 5: Comparison between the hand calculation and the RADTRAD model

Scenario Case	TEDE(Total Effective Dose Equivalent) : mSv		
	Hand Calculation	New methodology (RADTRAD)	Current methodology (RADTRAD)
Normal	6.650E-02	6.657E-02	6.641E-02
Re-loading	1.320E-01	1.321E-01	1.311E-01
Abnormal	1.570E-01	1.573E-01	1.565E-01

4 CONCLUSIONS

In the three scenarios of SFP cooling failure, radiological estimation by RADTRAD was carried out using a new methodology. As a result, the TEDE calculation result was within acceptance criteria. Also, the results using the conventional method and the new RADTRAD methodology were compared, and these methods were found to be in good agreement.

5 REFERENCES

[1] IAEA, 2016. Safety of Nuclear Power Plants : Design, IAEA Specific Safety Requirements, No.SSR-2/1, Rev.1.

[2] KHNP co., ltd., Final Safety Analysis Report, chapter 9.1, 2003, rev.1.

[3] Cothern, C.R., Smith, Jr, J.E., 1987. Environmental Radon. Plenum Press, New York, pp. 98–107.

[4] Cheon, B., Yoo, D., Shin, W., et al., et al., 2019. Development of advanced skin dose evaluation technique using a tetrahedral-mesh phantom in external beam radiotherapy: a Monte Carlo simulation study. Physics in Medicine and Biology, 64(16), 165005. https://doi.org/10.1088/1361-6560/ab2ef5.

[5] Zölzer, F. (2012). A cross-cultural approach to questions of ethics in radiation protection. IRPA 13 Glasgow presentation. Retrieved 2013-01-31 from: http://s281354445.websitehome.co.uk/Glasgow/Projects/IRPA2012/NewWebsite/wp-content/uploads/2012/06/Presentations/Dochart/thu/0940%20thu%20dochart%20Zolzer.ppt.

[6] US NRC, NUREG-6604, RADTRAD: A Simplified Model for RADionuclide Transport and Removal And Dose Estimation, SAND98-0272, Osti.gov, 1998