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Development of Methodology for Spent Fuel Pool Severe Accident Analysis Using MELCOR Program

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1. Introduction

The general reason why SFP severe accident analysis has to be considered is that there is a potential great risk due to the huge number of fuel assemblies and no containment in a SFP building. In most cases, the SFP building is vulnerable to external damage or attack. In contrary, low decay heat of fuel assemblies may make the accident processes slow compared to the accident in reactor core because of a great deal of water. In short, its severity of consequence cannot exclude the consideration of SFP risk management.

The U.S. Nuclear Regulatory Commission has performed the consequence studies of postulated spent fuel pool accident [1]. Pertinent research conducted over the last several decades is summarized in NUREG-1353, "Regulatory Analysis for the Resolution of Generic Issue 82, Beyond Design Basis Accidents in Spent Fuel Pools, April 1989; in NUREG/CR-6451, "A Safety and Regulatory Assessment of Generic BWR and PWR Permanently Shutdown Nuclear Power Plants," April 1997 and in NUREG-1738, "Technical Study of Spent Fuel Pool Accident Risk at Decommissioning Nuclear Power Plants," February 2001.

Post the Fukushima nuclear disaster in 2011, much attention has been paid to investigation of severe accidents (SA) progression in spent fuel pool (SFP) of nuclear power plants (NPP). The Fukushima-Daiichi accident has accelerated the needs for the consequence studies of postulated spent fuel pool accidents, causing the nuclear industry and regulatory bodies to re-examine several assumptions concerning beyond-design basis events such as a station blackout. The tsunami brought about the loss of coolant accident, leading to the explosion of hydrogen in the SFP building.

Analyses of SFP accident processes in the case of a loss of coolant with no heat removal have studied [2,3,4,5]. Few studies however have focused on a long-term process of SFP severe accident under no mitigation action such as a water makeup to SFP. USNRC and OECD have co-worked to examine the behavior of PWR fuel assemblies under severe accident conditions in a spent fuel rack. In support of the investigation, several new features of MELCOR model have been added to simulate both BWR fuel assembly and PWR 17x17 assembly in a spent fuel pool rack undergoing severe accident conditions [6].

The purpose of the study in this paper is to develop a methodology of the long-term analysis for the plant-level SFP severe accident by using the new-featured MELCOR program in the OPR-1000 Nuclear Power Plant. The study is to investigate the ability of MELCOR in predicting an entire process of SFP severe accident phenomena including the molten corium and concrete reaction. The framework of the MELCOR analysis will be helpful to establish a basis for the severe accident mitigation strategy.

2. Methodology

2.1 Selection of Scenarios

For the severe accident analyses of spent fuel pool by using MELCOR code, the two scenarios to be considered in this study are a loss of cooling accident and a loss of coolant accident in order to determine the modeling capabilities of MELCOR. The loss of cooling accident is initiated by assuming only the complete failure of cooling function in SFP cooling system. This scenario might be the most possible severe accident because of a robust structure of the spent fuel pool. The loss of coolant accident with a loss of heat removal, however, should be analyzed in the point of view of its severity. New SFP features [7] in MELCOR 1.8.6 (Rev.03) have capabilities that can analyze a wide range of postulated SFP accident scenarios. These new SFP features can be used to perform two types of SFP evaluations: normal operation conditions, partial loss of coolant inventory accident.

Table 1 Selection of Accident Scenarios

	Accident Scenarios	Operation Mode	Remarks
Case 1	Loss of Cooling Accident	Normal	Initially normal water level
Case 2	Loss of Coolant Accident	Normal	

2.2 MELCOR Modeling Consideration

2.2.1 Status of Spent Fuel Storage

Spent fuel pools are storage pools for the spent fuel assemblies withdrawal from nuclear reactor vessel. The dimensions in OPR-1000 NPPs, are typically 40 x 35 x 28 feet in depth, width, and length, respectively. Figure 1 shows a 3-dimensional SFP configuration. In recent, licensee has replaced the original storage racks in the spent fuel pool with high density racks in which neutron absorber plates are attached in order to increase spent fuel storage. And about 60% of the racks are stored with spent fuel assemblies so far. Since the storage capacity causes a higher risk of consequence due to the increased amount of radioactive materials, the importance of the risk management becomes higher in the SFP building.

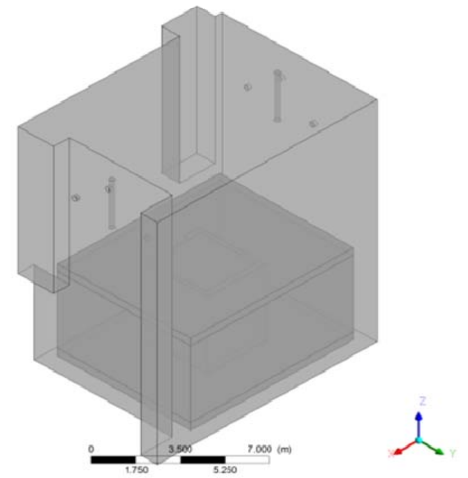


Figure 1 3-Dimensional SFP Configuration

2.2.2 Spent Fuel Heat Load Calculation

The decay heat load generated by the spent fuel is calculated based on the dimensionless fractional power. The previous decay heat load from the existing spent fuels that had been stored before the last withdrawal was calculated by using Eq.(1). The total decay heat loads including the last withdrawal spent fuels were calculated by Eq.(2) for the plant normal operation.

$$P_{cons} = \frac{68}{177}(\sum(W \cdot H51)) \quad (1)$$

$$Q_n = \frac{68}{177} \cdot W \cdot H51 + P_{cons} \quad (2)$$

where:

P_{cons}: Previous heat load (MWt)
Q_n: Heat load newly withdrawal in normal operation (MWt)
W: Core thermal power (MWt)
H51: Dimensionless fractional power of 51 GWD/MTU burn-up

Table 2 shows the number of fuel assemblies withdrawal from reactor vessel to SFP and their heat load. Start time of withdrawal in normal mode was assumed to be 100 hours after a reactor trip in the heat load calculation.

Table 2 Summary of Heat Load Calculations

Withdrawal histories	Operation Mode	Number of Fuel Assembly (EA)	Heat Load (MWt)
Newly withdrawal	Normal	68	4.1
Previous	-	860	1.1

3. Development of MELCOR Input Model

In order to analyze the severe accident scenarios of spent fuel pool, the following MELCOR packages [8] were mainly used to model spent fuels, racks, pool, and SFP building.

3.1 COR Package Modeling

MELCOR 1.8.6 code enhancement to SFP modeling was carried out by adding two new features to the COR package: (1) a new rack component, which permits modeling of a SFP racks, and (2) an enhanced air oxidation kinetics model [7]. The former allows separate modeling of the SFP rack and radiation heat transfer between fuel and rack. The later evaluates the transition to breakaway oxidation kinetics in air environments.

Spent fuel and racks modeling using MELCOR COR Package was performed reflecting the characteristics of SFP structure and heat loads of spent fuel as shown in Fig. 2. In the case of normal operation mode, 68 newly-withdrawal fuel assemblies were modeled as Ring 1 and their heat load is applied as 4.1 MWt. 860 previous fuel assemblies released from reactor vessel were modeled as Ring 2 and its heat load was applied to be 1.1 MWt. The empty storage racks (without spent fuel) were modeled as Ring 3.

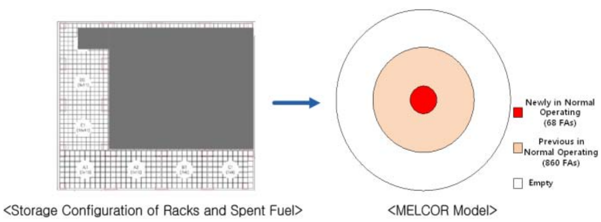


Figure 2 Modeling of Spent Fuel Assemblies in COR Package

3.2 CV/FL Package Modeling

The pool walls can be modeled using heat structure (HS package) and the pool water inventory and SFP building are modeled using the CVH package. Thirty seven (37) control volumes and thirty nine (39) flow paths using CV package and FL package are listed in Table 3. Figure 3 shows the nodalization of control volumes and flow paths. It was assumed that a leak path to atmosphere was formed through stair way as shown in Fig. 3.

Table 3 CV and FL Modeling for Geometric Structure

	CVH Package	FL Package
Ring 1	CV101 ~ CV109	FL101 ~ FL110
Ring 2	CV201 ~ CV209	FL201 ~ FL210
Ring 3	CV301 ~ CV309	FL301 ~ FL310
Upper volume of SFP building	CV020	FL020, FL021, FL022
Stair rooms	CV021 ~ CV023	FL002, FL005
Environment	CV003	FL023

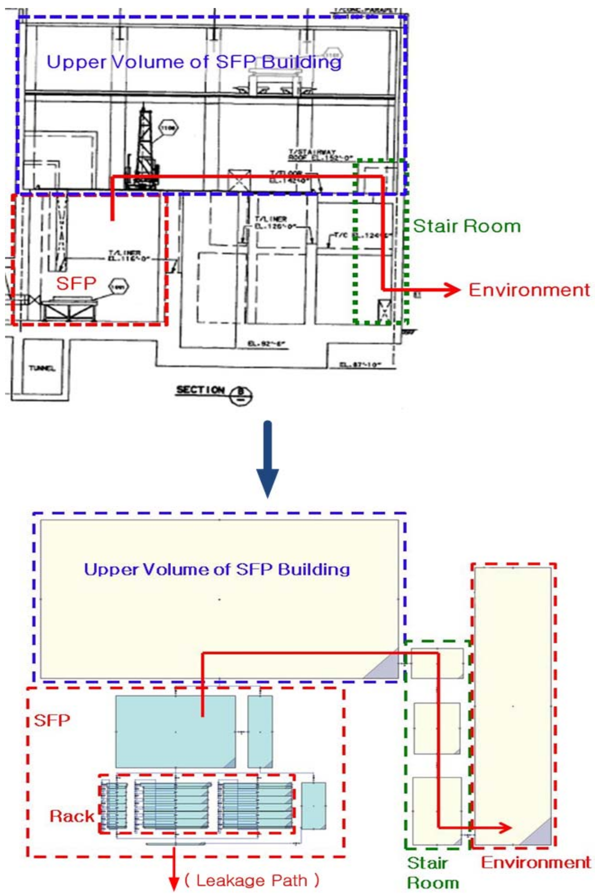


Figure 3 Nodalization of SFP and Building

3.3 CAV Package Modeling

In order to predict the phenomena of molten fuel concrete interaction after spent fuel collapse to pool bottom, the modeling of CAV package were performed as shown in Fig. 4. The relocated fuel assembly and core melting product are expected to move to the bottom of spent fuel pool. This behavior was analyzed by using lower head model for reactor vessel. However, because spent fuel pool does not have the structure such as lower head of reactor vessel, the hypothetical lower vessel head was considered. It is mentioned in NUREG/CR-6119 [7] that the lower head should be modeled as a flat plat of user-specified composition and should have a uniform thickness even at the junction with the edge of the rack. The COR package is hardwired with CVH and CAV packages to include a CVH volume below the lower head to represent the cavity volume below the core vessel.

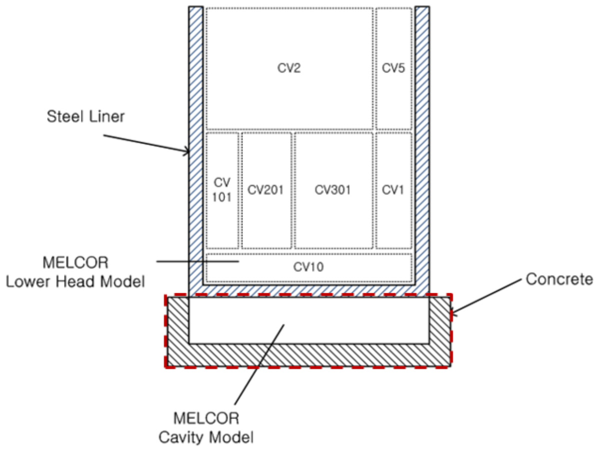


Figure 4 SFP Pool and Cavity Modeling

4. MELCOR Analyses

The severe accident analyses for two SFP accident scenarios were performed by using SFP MELCOR input model developed in the present study as summarized in Table 1.
The major results of analysis are described with respect to a level of coolant, cladding temperature and mass of hydrogen generation. The duration of accident progression was applied to be 240 hours (10 days).

4.1 Loss of Cooling Accident

As shown in Fig. 5(a), after the loss of cooling function occurred, the level of coolant, because of coolant boiling, continuously decreased and then the spent fuel rack was uncovered at about 95 hours, the spent fuel was uncovered at about 100 hours. Eventually, the pool was completely empty at about 157 hours.

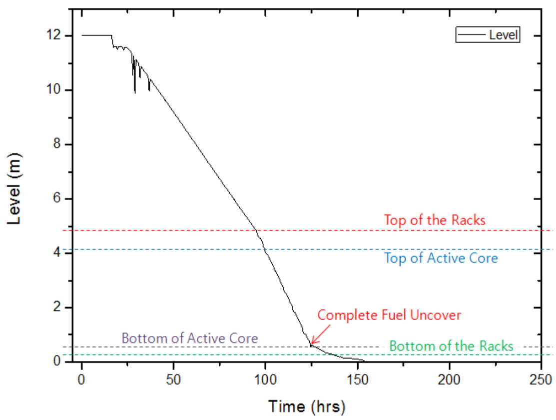
The cladding temperatures in Ring 1 were started to increase sharply at 125 hours as depicted in Fig. 5(b). The rapid rise of cladding temperatures in Ring 2 were started at a similar time and showed a slower increment compared to the case of Ring 1. Spent fuel claddings in

Ring 1 were failed at 143 hours and in Ring 2 at 164 hours, leading fuel assembly degradation after starting with Zr ignition.

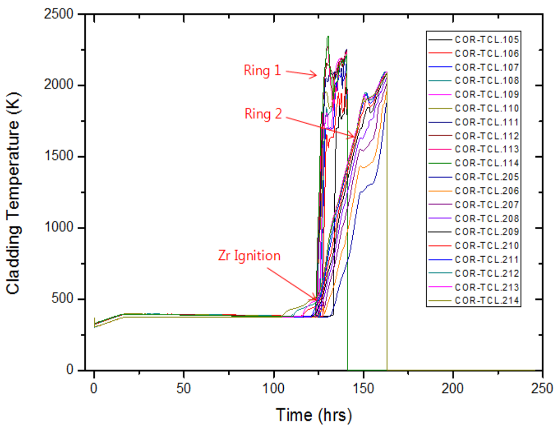
The accumulated hydrogen masses generated by Zr oxidation and MCCI (Molten Core-Concrete Interaction) were, in Fig. 5(c), 221 kg and 3,314 kg, respectively.

Table 4 Sequence in Normal Mode for Loss of Cooling Accident

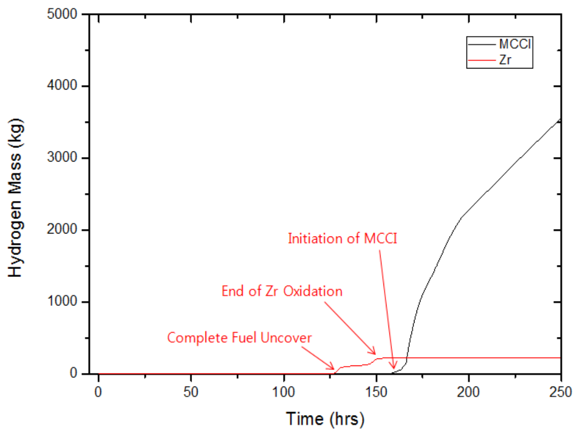
Time (hrs)	Events
0	Loss of Cooling
95	Start of SFP Rack Uncover
125	Rapid Temperature Rise in Cladding
143	Collapse of Ring 1
157	Complete loss of inventory
159	Corium relocation to bottom of SFP
159	Start of Hydrogen by MCCI
164	Collapse of Ring 2
240	1 m-deep Concrete Ablation



(a) Water level



(b) Cladding temperature



(c) Hydrogen generation

Figure 5 Analysis Results for Loss of Cooling Accident

4.2 Loss of Coolant Accident

According to NEI 06-12 [10], it was reported that:

> "If the area around the spent fuel pool is accessible, then a determination of the spent fuel pool leakage rate should be made. This determination should focus on the relative rate of loss of inventory is excessive (i.e., does pool level indicate that the leak rate is likely greater than 500 gpm, or is dose rate excessive due to fuel uncover). If it can be determined that the leakage rate is not excessive, then makeup should be initiated using the internal strategy, supplemented by the external makeup strategy, as necessary to maintain or restore water level."

For the loss of coolant accident, 2-inch diameter opening at the bottom of SFP was postulated. As shown in Fig. 6, the leak rate was calculated to be 38 kg/sec (about 600 gpm) at a maximum. As the water level decreased, the leak rate decreased empting the SFP at about 15 hours.

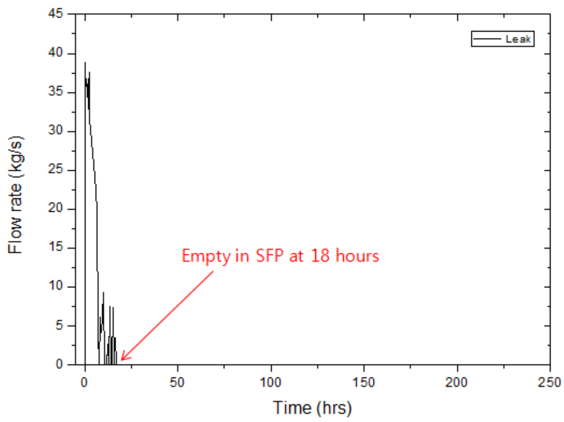
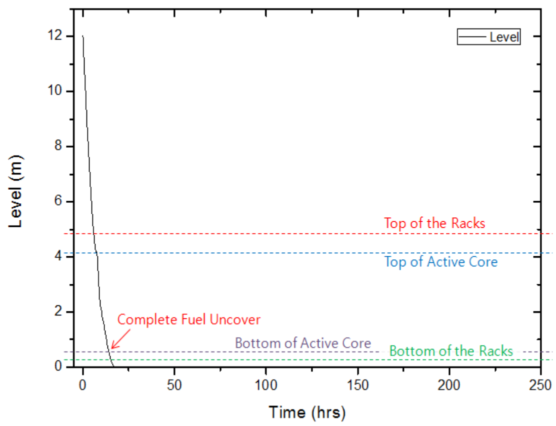


Figure 6 Leakage Rate through SFP Bottom

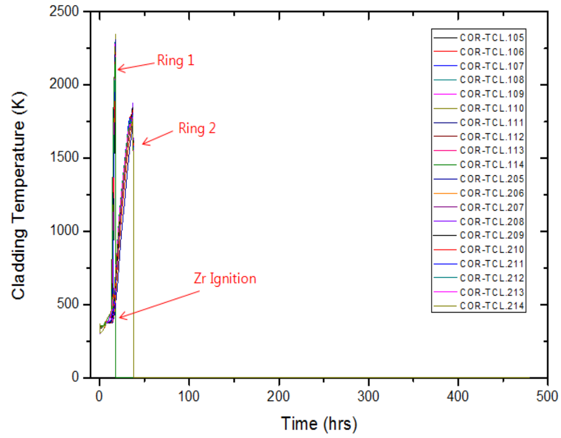
As shown in Fig. 7(a), after the loss of coolant occurred without heat removal function of the SFP, the level of coolant, because of the postulated leak, continuously decreased and then the spent fuel rack was rapidly uncovered at about 6 hours, the spent fuel was uncovered at about 7 hours. Eventually, the pool was completely empty at about 18 hours.

The cladding temperatures in Ring 1 were started to increase sharply at 14 hours as depicted in Fig. 7(b). The rapid rise of cladding temperatures in Ring 2 were started at a similar time and showed a slower increment compared to the case of Ring 1. Spent fuel claddings in Ring 1 were failed at 18 hours and in Ring 2 at 37 hours, leading fuel assembly degradation after starting with Zr ignition.

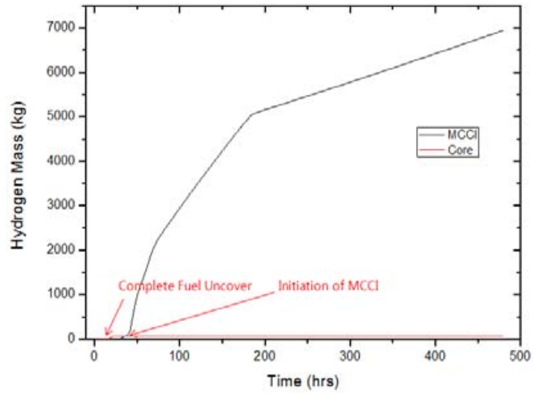
In Fig. 7(c), the accumulated hydrogen masses generated by Zr oxidation and MCCI were 64 kg and 5,405 kg, respectively.



(a) Water level



(b) Cladding temperature



(c) Hydrogen generation

Figure 7 Analysis Results for Loss of Coolant Accident

5. Summary and Conclusions

As shown in Table 5, all postulated major scenarios were estimated resulting in the reduction of water inventory level and the mass of hydrogen generation. In the case of loss of cooling accident, time to water level reduction was obviously dependent of the decay heat load. It was revealed that the loss of inventory was dominant factor in the case of loss of coolant accident. With respect to the mass of hydrogen generation, massive amount of hydrogen was generated from the molten corium concrete reaction in MELCOR calculation.

Table 5 Summary of MELCOR Analyses Results

Accident Scenarios	Water Level Reduction Time (hr)	Hydrogen Generation (kg)	
	Uncover	Cladding Oxidation	MCCI
Loss of Cooling Accident	95	221	3314
Loss of Coolant Accident	6	64	5405

Through the MELCOR modeling developed in this study, it was found that MELCOR program has capabilities to predict entire progress of water level, fuel degradation, hydrogen generation, and concrete reaction. This framework of SFP severe accident analysis method will be useful in determining operation actions for the accident mitigation procedures. As the present study has been conducted, it is found however that further studies are needed to complete the methodology of SFP severe accident analysis as follows:

- Feasibility study of fuel degradation inside racks,
- Investigation of uncertainties of MCCI,
- Enhancement of SFP building modeling to find a critical leak path, and
- Completion of framework to accident mitigation strategy.

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