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THE PLANT-SPECIFIC SPENT FUEL POOL SEVERE ACCIDENT ANALYSIS TO SUPPORT A SFP RISK AND ACCIDENT MANAGEMENT

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ABSTRACT

In recent times, the main concerns related to the spent fuel pool (SFP) of a nuclear power plant (NPP) have been focused on (a) controlling the configuration of the fuel assemblies in the pool with no loss of pool coolants, and (b) ensuring adequate pool storage space to prevent fuel criticality and keep the fuel cool from the perspective of a design basis accident (DBA). The Fukushima accident on March 11, 2011, however, has raised the possibility of severe accidents in an SFP under beyond design basis external events (BDBEEs) that might lead to a loss of all safety functions and need a way to cope effectively with such an event through a SFP risk and accident management. From a SFP safety point of view, the low decay heat of the fuel assemblies and large water inventory in an SFP may make the accident processes slow compared to an accident in the reactor core, but a huge number of fuel assemblies stored inside it and no containment in a SFP building (especially for PWR types) might be exposed to much greater risk. A quantitative analysis for SFP accidents can give more insights into which aspects play dominant roles in an accident progression. To date, however, few studies have been made for severe plant-level SFP accidents. This paper provides the plant-specific analysis results for severe accidents that are expected in the SFP of a typical PWR, which has been performed using a severe accident analysis code, MELCOR1.8.6.

INTRODUCTION

Effective management of Spent Fuel Pool (SFP) safety has been raised as one of emerging issues to further enhance nuclear installation safety after the Fukushima accident on March 11, 2011. Before then, the SFP safety-related issues have been mainly focused on (a) controlling the configuration of the fuel assemblies in the pool with no loss of pool coolants, and (b) ensuring adequate pool storage space to prevent fuel criticality owing to chain reactions of the fission products and the ability for neutron absorption to keep the fuel cool. The safety and risk assessment for beyond design basis accidents (BDBAs) in SFP greatly helps (a) identify which SFP equipment is of importance to the risk aspect and (b) which equipment and procedures should be primarily modified to enhance the overall SFP risk profile. In the US, the resolution of GI-82 based on the NRC Safety Goal Policy Statement (Throm, 1989) requires the contribution of SFP accidents to be *a few percent* of the overall CDF (Core Damage Frequency) target of 1.0×10^{-4}/ry.

The risk of BDBAs in SFP examined first in WASH-1400 (USNRC, 1975) was orders of magnitude below those involving the reactor core because of the simplicity of the SFP:

- The coolant in SFP is at atmospheric pressure;
- The spent fuel is always subcritical and the heat is low;
- There is no piping that can drain the pool water;
- There are no anticipated operational transients that could interrupt the cooling or cause a criticality.

Thereafter, many studies on SFP safety have been reflecting the following circumstances:

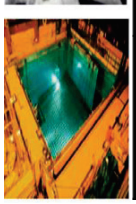
- Spent fuel is being stored instead of reprocessed, leading to an expansion of onsite fuel storage by means of high-density storage racks, which results in a larger inventory of fission products in the pool and a greater load on the pool cooling system;
- Some additional studies have provided evidence of the possibility of fire propagation between the assemblies in an air-cooled environment (Benjamin et al., 1979; Sailer et al., 1987; Durbin et al., 2012 & 2013);
- Fuel assemblies in SFP may be severely damaged during a fuel crud removal operation (OECD, 2008).

After the 9/11 attacks in the US in 2001, SFP safety researches have focused more on (a) assessing the effectiveness of mitigation measures on SFP, (b) improving the analysis of fuel coolability and heatup, (c) improving the fuel configuration within the pool to achieve a substantially greater passive cooling capability through natural convection, and (d) further understanding and validating the analytical modelling through relevant experiments. The Fukushima nuclear disaster has stimulated the need for in-depth research on SFP safety and the related regulation requirements.

The first comprehensive analysis on SFP risk was made through the NUREG-1738 study (USNRC, 2001). The SFP risk assessed in this study was quite low owing to the low frequency of events that could damage the thick reinforced pool walls: frequency of the fuel uncovery, $1.0\text{x}10^{-7}$ to $1.0\text{x}10^{-6}$/ry. However, the foregoing consequences have been assessed to be relatively large, by compounding overly conservative estimates of seismic risk, pool fragility and the probability and magnitude of the postulated zirconium fire and its consequent releases, and the large inventory of Cs-137 in SFP. Nevertheless, the foregoing study provided valuable insight from the aspect of key contributors to the generic SFP risk profile, including the following:

- Decay heat of spent fuels stored in SFP,
- Likelihood of draining the spent fuel pool given realistic seismic events,
- Likely configuration of fuel following an event that could drain the pool,
- Likelihood of cladding oxidation propagating beyond assemblies with the highest decay heat,
- Time period over which postulated releases could occur, and
- Recovery actions available to eliminate or mitigate potential releases.

Table 1: A framework for SFP risk assessment.



SFP-Level 1 PSA	SFP-Level 2 PSA
• **Major IEs Analysis** - Loss of cooling accident (**LOCA**) - Loss of pool inventory (**LOPI**) - Loss of offsite power (**LOOP**) - Station blackout (**SBO**) - **Cask drop** caused by human error, - **SFP structural failure**, due to seismic events (including concurrent tsunami) - **Internal fire**, due to pool structural failure and seismic initiator, cask drop, etc. - **Security-related Event**: Man-made external attack such as aircraft impact, external fire and explosion	• **SA Phenomena in SFP** - **Spent fuel rods** (fuel heatup and uncovery, severe damage, zirconium oxidation/ ignition, fuel melting, etc.) - **Spent fuel assembly** (heat transfer, fire propagation, fuel assembly collapse, etc.) - **Spent fuel storage rack and downcomer** next to the edge of the pool - **Base region beneath the racks** (cooing air ingression into the fuel assembly, MCCI, Basemat melt-through etc.) - **Spent fuel storage buildings** (hydrogen burn, air cooling etc.) - Effect of **AM strategies**
• **System and Accident Sequence Analysis** - **Event Tree/Fault Tree (ET/FT)** Analysis - Level 1 risk surrogate metric: Spent **Fuel Damage Frequency (FDF)**	• **Accident Progression and Source Term Releases Analysis** - APET Modeling and Analysis - Level 2 risk surrogate metric: **Large Release Frequency (LRF)**

Although the underlying accident phenomena and progressions in an SFP are somewhat different from the reactor case subject to high pressure, temperature, and configuration of the fuels, a similar framework to assess an accidental risk can be formulated by employing three consecutive levels of PSA (Probabilistic Safety Assessment) that have been applied to reactor cases. Table 1 summarizes a basic framework for the SFP risk assessment that can be performed as a part of an integrated risk assessment and the key technical elements.

The present study has been performed to support the SFP risk and accident management, the scope of which covers plant-specific analyses of severe accident phenomena (listed in SFP-Level 2 PSA of Table 1) for two representative SFP initiating events (LOCA and LOPI) that can be expected in a typical PWR SFP. A computational tool for severe accidents, MELCOR1.8.6 (Gauntt, 2005), was used for this purpose.

ANALYSIS METHODOLOGY

Reference plant

The OPR1000 (Korean Standardized NPP) (KEPCO, 1996) was selected as the reference plant for the present study, which is a two-loop pressurized water reactor with a capacity of 1,000 MWe (a rated thermal power of 2,815 MW_{th}) and with a large-dry containment. The SFP is located adjacent to the primary containment (outside primary containment), and its bottom is placed above the plant grade. The number of fuel assemblies that can be stored inside the SFP is about 1,770. The squared storage racks are made of stainless steel and there is no cross flow between rack cells. Figure 1 shows a 3-dimensional configuration of the OPR1000 SFP, which is 40 x 35 x 28 feet in depth, width, and length, respectively.

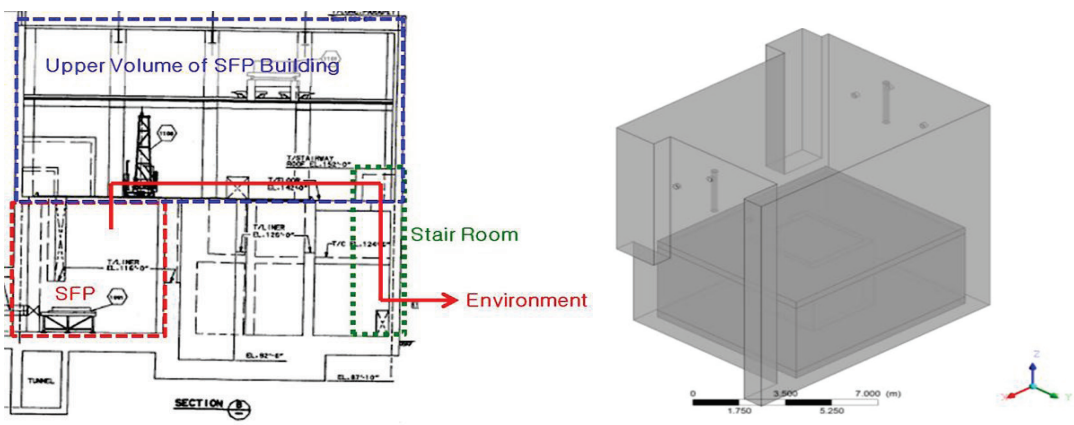


Figure 1. Layout of SFP of the reference plant and its 3-dimensional configuration

Accident Scenarios

A loss of water from the SFP can occur as a result of water evaporation in a slowly evolving loss of cooling scenarios or water leakage owing to damage to the pool structures (including pipe penetrations). Table 1 provides two representative accident scenarios selected for the present analysis of severe accidents that are expected in the SFP: (a) a loss of cooling accident (LOCA) and (b) a loss of pool inventory (LOPI). The loss of cooling accident is typically induced from a complete loss of SFP safety/cooling systems. LOCA has been considered as the most likely event for the SFP due to the robust structure of the spent fuel pool that reduces the likelihood of LOPI. The LOPI, which leads to a loss of cooling water, can be caused by a (seismic-induced) break of the SFP wall or penetration pipes. This

event should be analyzed by taking into account a spectrum of the break sizes and locations, for which different impacts are anticipated.

Table 2: Two representative accident scenarios for analysis.

	Accident Scenarios	Operation Mode	Remarks
Case 1	LOCA: Loss of Cooling Accident	Normal	Initially normal water level
Case 2	LOPI: Loss of Pool Inventory	Normal	

Heat Loads from Spent Fuels

Because progression of severe accidents evolves from a loss of heat and mass balance, the decay heat load being radiated from spent fuels is an essential part of an SFP accident analysis. To estimate the decay heat loads from FAs in the SFP, the present study utilizes the dimensionless fractional power. The decay heat load from the existing spent fuels that had been stored in the SFP before the last withdrawal (offload) from the reactor core is calculated using Eq. (1). The total decay heat load including the last withdrawn spent fuels can then be calculated by Eq. (2) for the case of a normal operation.

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

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

where

P_{cons}: Heat load from the previous fuel assemblies (FAs) stored in SFP (MW_{th})
Q_n: Heat load from newly withdrawn FAs during a normal operation (MW_{th})
W: Thermal power of the reactor core (MW_{th})
H51: Dimensionless fractional power of an assumed fuel burn-up (51 GWD/MTU)

Table 3 shows the maximum number of FAs that can be withdrawn from the reactor core to the SFP per one fuel cycle (18 months in case of OPR1000) and decay heat load estimated from those FAs. In normal operation mode, the last withdrawal of FAs from the reactor core is assumed to be 100 hours after a reactor trip, which is the maximum time allowed to move the spent fuels to SFP in case of OPR1000.

Table 3: Heat loads according to the withdrawal history of the fuels.

Withdrawal (offload) histories	Plant operation mode	Number of fuel assemblies (FA)	Decay heat load (MW_{th})
100 hours after offload	Normal	68	4.1
Previously stored in SFP		860	1.1

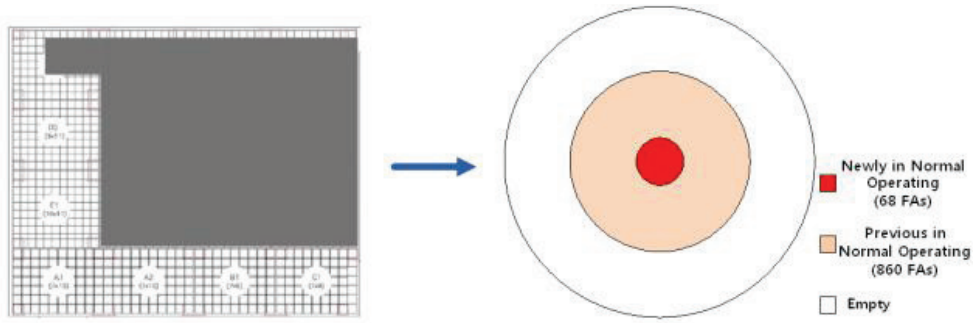
MELCOR Modelling Inputs

To analyze the key severe accident phenomena expected in the SFP, the plant-specific MELCOR inputs were developed for spent fuels, racks, pool, and SFP building.

Modelling of COR Packages

To enhance the modelling capability of the SFP, two new features have been recently added to the MELCOR 1.8.6 COR package: (a) a new rack component, which permits modeling the SFP racks, and (b) an enhanced air oxidation kinetics model (Gauntt, 2005). The former allows a separate modeling of the

SFP rack and radiation heat transfer between the fuel and rack. The latter evaluates a transition to the breakaway oxidation kinetics in air environments. Fig. 2 shows the modeling of the fuel assemblies and racks through the MELCOR COR package, reflecting the plant-specific SFP structure and spatial distribution of FAs and racks. In the case of normal operation mode, Ring 1 consists of 68 newly-withdrawn fuel assemblies, with the decay heat load estimated to be 4.1 MW_{th}. Ring 2 models 860 previous fuel assemblies released from the reactor core, with an applied heat load of 1.1 MW_{th}. The empty storage racks (without spent fuel and heat load) were modeled as Ring 3.



(a) Spatial distribution of FAs and racks (b) MELCOR model of FAs

Figure 2. Modeling of fuel assemblies in COR package.

Modelling of CVH and FL Packages

The SFP pool wall was modeled using a heat structure (HS package) and the pool water inventory and SFP building were modeled through the CVH package. Table 4 shows the present modeling for the CV package (using 37 control volumes) and FL package (using 39 flow paths). Fig. 3 shows the corresponding computational nodes for the control volumes and flow paths for SFP and building as well. A leak path to the atmosphere was assumed to form through a stair way.

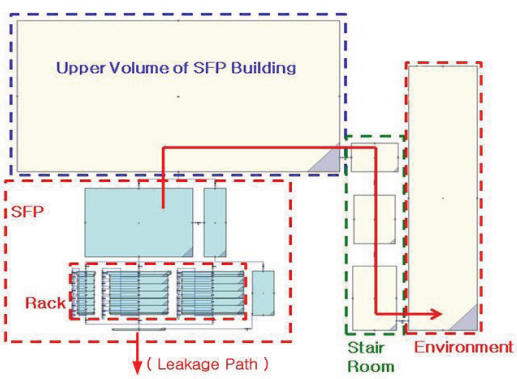


Figure 3. Nodalization of SFP and building.

Table 4: CVH/FL modeling for SFP and building.

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

Modelling of CAV Package

Fig. 4 shows a modeling of the CAV package to predict molten fuel concrete interaction (MCCI) phenomena, which could happen after a collapse of FAs to the floor of the SFP. The relocated fuel assembly and (damaged and/or melted) fuel debris moving to the bottom of the SFP were analyzed through a reactor vessel lower head model of the MELCOR code. However, because the SFP geometry is not the same as the lower head of the reactor vessel, the hypothetical lower vessel head was considered according to a recommendation of NUREG/CR-6119 (Gauntt, 2005) in which the lower head should be modeled as a flat plate of a 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.

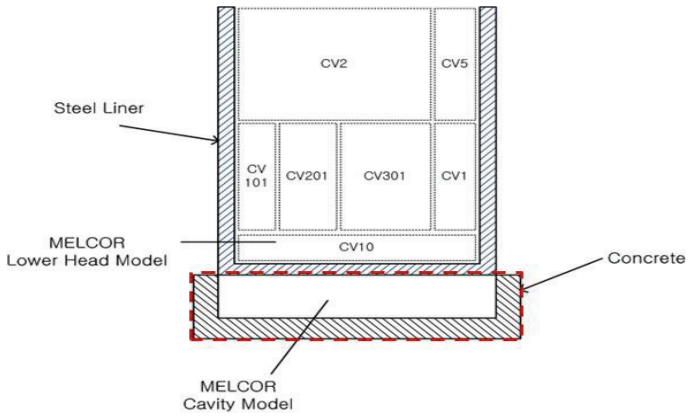


Figure 4. SFP pool and cavity model.

ANALYSIS RESULTS

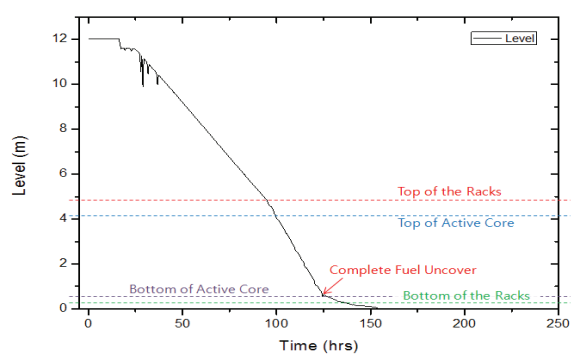
For the two SFP accident scenarios in Table 2, the plant-specific severe accident analyses were performed based on the foregoing MELCOR input models. To catch up with the key accident phenomena ranging from a time to fuel uncover to MCCI (Molten Corium Concrete Interaction), a MELCOR simulation was made for up to 240 hours (10 days). Major results are described with respect to the level of the coolant, the cladding temperature and the mass of hydrogen generated during accident progressions.

Loss of Cooling Accident (LOCA)

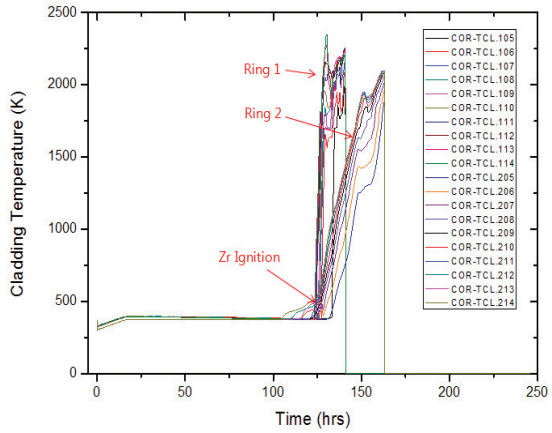
Fig. 5(a) shows that after a loss of cooling function occurs, a pool boil-off is progressed; the level of coolant continuously decreases, and the top of the spent fuel rack is then exposed at about 95 hours, and the top of the spent fuel is uncovered at about 100 hours, and eventually the pool is completely dried out at about 157 hours. Fig. 5(b) shows that the cladding temperatures in Ring 1 begin to increase sharply at 125 hours. The cladding temperatures in Ring 2 begin to rapidly increase at a similar time, but at a slower rate compared to the case of Ring 1. The fuel claddings in Ring 1 and Ring 2 fail at 143 hours and 164 hours, respectively, leading to a further degradation of FAs after a start of Zirconium ignition. MCCI happens just after Ring 2 fails and accordingly generates non-condensable gases containing hydrogen. Fig. 5(c) and Fig. 6(d) show that burnable hydrogen of 221 kg and 3,314 kg was generated by Zirconium oxidation and MCCI, respectively, until the end of the calculation.

Table 5: LOCA: Key Events in Normal Mode

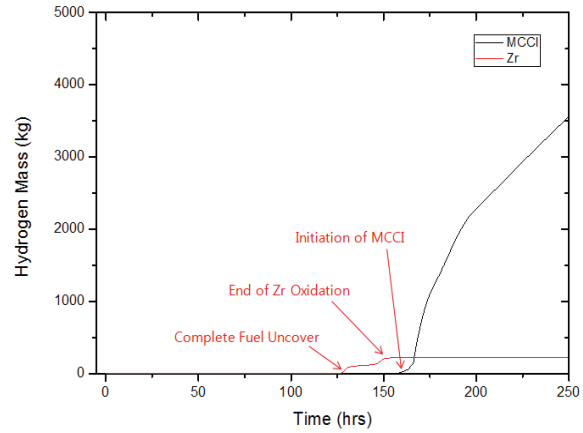
Time(hrs)	Chronological Events
0	Loss of cooling
95	Start of SFP rack uncover
125	Rapid temperature rise in cladding
143	Collapse of Ring 1
157	Complete loss of pool inventory
159	Corium relocation to bottom of SFP
159	Generation of hydrogen by MCCI
164	Collapse of Ring 2
240	1 m-deep concrete ablation



(a) SFP water level



(b) Cladding temperature



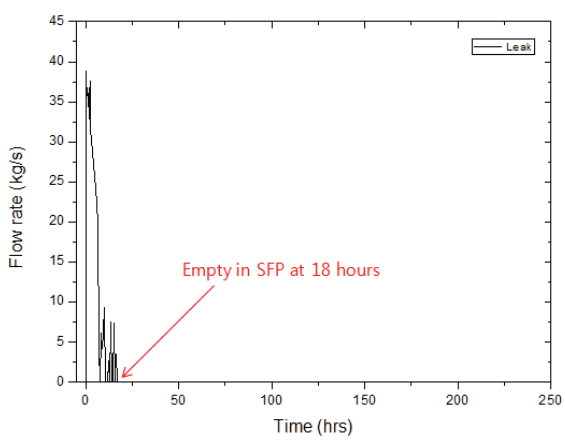
(c) Hydrogen generation

Figure 5. MELCOR analysis results for LOCA

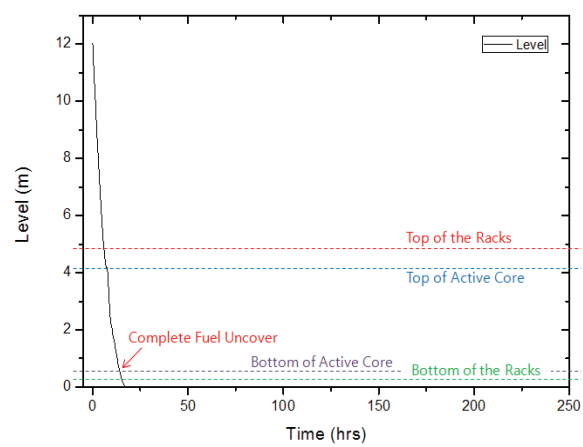
Loss of Pool Inventory (LOPI)

NEI 06-12 (NEI, 2006) points out 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 the makeup should be initiated using the internal strategy, supplemented by the external makeup strategy, as necessary to maintain or restore water level."

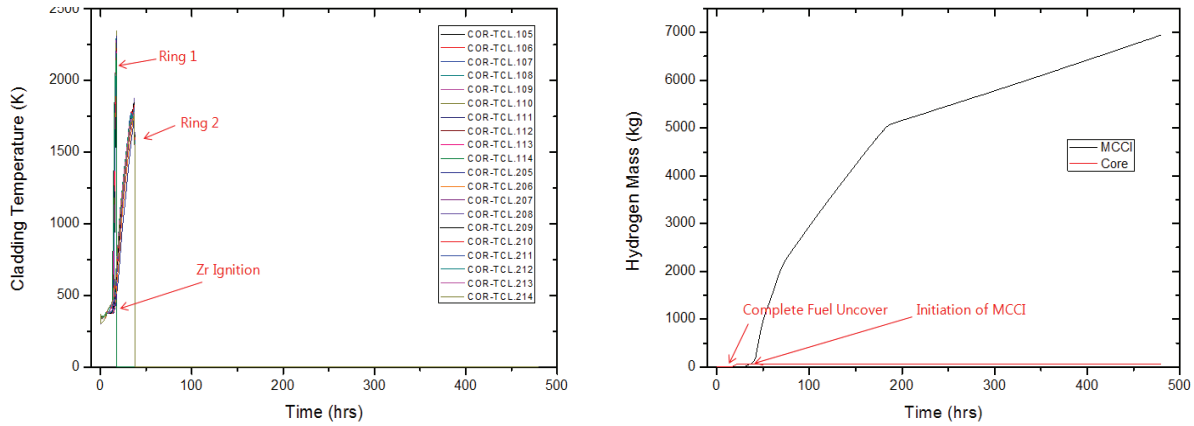
The reason why the above statement is mentioned here is because we need plant-specific information on the size and location spectrum of a seismic-induced SFP wall or pipe breaks, which are essential for the LOPI-induced accident analysis, but we currently do not have any information about it. Thus, a 2-inch diameter opening at the bottom of SFP was postulated for the present loss of coolant accident analysis. Fig. 6(a) shows that the break size induces a maximum leak rate of 38 kg/sec (about 600 gpm), and as the water level decreased the leak rate decreased, drying out the SFP at about 15 hours.



(a) Leak rate through SFP bottom



(b) SFP water level



(c) Cladding temperature (d) Hydrogen generation

Figure 6. MELCOR analysis results for LOPI

Fig. 6(b) shows that just after a loss of coolant accident occurs without the heat removal function of the SFP, the level of the coolant rapidly decreases, leading to an uncover of the spent fuel rack at about 6 hours followed by uncovering of the spent fuel at about 7 hours. The pool is completely dried out at about 18 hours. As depicted in Fig. 6(c), the cladding temperatures in Ring 1 begin to increase sharply at 14 hours. The rapid rise of the cladding temperatures in Ring 2 is made at a similar time, but at a slower rate compared to the case of Ring 1. Fuel claddings in Ring 1 failed at 18 hours and at 37 hours in the case of Ring 2, leading to a further degradation of FAs after a start of a Zirconium fire ignition. Fig. 6(d) shows that burnable hydrogen of 64 kg and 5,405 kg was generated by Zirconium oxidation and MCCI, respectively, until the end of the calculation.

SUMMARY AND CONCLUSION

To obtain key insights into the underlying severe accident phenomena expected in an SFP, plant-specific analyses for two potential accident scenarios have been performed using the MELCOR code. For this purpose, the plant-specific MELCOR inputs were developed for spent fuels, racks, pool, and SFP building. Table 6 summarizes the key accident phenomena obtained through the present analysis. In the case of a loss of cooling accident, the time to the pool uncover is obviously dependent on the level of decay heat load. In the case of a loss of pool inventory, the break size and location could be a dominant factor in determining the times to the key phenomena and further accident progression.

Table 6: A summary of the present MELCOR analysis results

Accident Scenarios	SFP water level and inventory		Hydrogen generation (kg)	
	Time to pool uncover (hrs)	Time to pool dry-out (hrs)	Cladding oxidation	MCCI
LOCA	95	155	221	3,314 (concrete ablation: 0.97 m)
LOPI	6	18	64	5,405 (concrete ablation: 1.47 m)

For both scenarios, a massive amount of hydrogen was generated from MCCI. The foregoing results provide the very important insights from the point of SFP risk and accident management:

- Time to pool uncover can be used to estimate (a) the time available to take actions before any fuel

uncover and/or the time available before the zirconium fire after a fuel uncover, and (b) its contribution to FDF (Fuel Damage Frequency)

- Hydrogen generation can be used to predict (a) the possibility of hydrogen burn or detonation in the SFP building, the resulting pressure and mechanical loads to the SFP building, and (b) the contribution to the LRF (Large Release Frequency).

To obtain further insights into the plant-specific SFP risk and accident management, future studies should be more focused on investigating the following:

- Fuel degradation inside the SFP racks,
- Phenomenological uncertainties inherent in the MCCI,
- Further modelling of the SFP building to find a critical leak path, and
- Impacts of accident mitigation strategies on accident progression.

ACKNOWLEDGEMENT

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