# A Probabilistic Analysis Methodology and Its Application to A Spent Fuel Pool System

Hyowon Kim[a], Moosung Jae [a*], Ho G. Ryu[b]
[a]*Hanyang Univ., Haengdang-dong, Sungdong-goo, Seoul, Korea*
[b]*M&D Co., Daedeok R&D Center, 199 Techno Rd, Daejeon*
**Corresponding Author: jae@hanyang.ac.kr*

## 1. Introduction

After the Fukushima Accident, the safety in the spent fuel in the storage pool becomes one of the hot issues in nuclear power plants. The recent finding turns out that sampling spent fuel assemblies in the 4th NPP in the Fukushima Daiichi maintain the integrity. The potential risk in spent fuel pool system still exists. There was a similar accident occurring at the 2nd unit of PAKS nuclear power station in Hungary on the 10th April 2003. Insufficient cooling of spent fuel caused the spent fuel burn up or partly melting. There were many previous studies performed for analyzing and measuring the risk of spent fuel damage. In the 1980s, there are changes in conditions such as development of high density storage racks and new information concerning the possibility of cladding fires in the drained spent fuel pools. The USNRC assessed the spent fuel pool risk under the Generic Issue 82**.** In the 1990s, under the USNRC sponsorship, the risk assessment about the spent fuel pool at Susquehanna Steam Electric Station (SSES) has been performed and Analysis Evaluation of Operational Data (AEOD) has been organized for accumulating the reliability data.

## 2. Methods

A technique has been employed and applied for identifying the initiating events required for analyzing the probabilistic risk for Spent Fuel Pool (SFP). The Master Logic Diagram (MLD), Event Tree Analysis (ETA), Fault Tree Analysis (FTA) and other reliability analysis methods are introduced. Those methodologies are applied for a reference plant in Korea. The reference plant is designed by Westinghouse. The facility of the SFP is in the fuel building located outside the containment. The cooling system for the spent fuel pool has two trains. Each train has both a pump and a heat exchanger.

### *2.1 Master Logic Diagram*

Master Logic Diagram (MLD) is a method for identifying initiating events in nuclear power plants. The MLD is a logic diagram that resembles a fault tree but without the formal mathematical properties. By using the MLD for deriving initiating events in the SFP facility, three main topics are selected; they are 'Inadequate Heat Removal', 'Loss of Inventory' and 'Excessive Heat Load. The MLD for the SFP facility is developed as shown in Fig. 1 below.

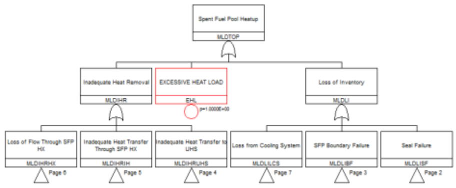


Fig. 1. Master Logic Diagram for a SFP facility.

They might include the loss of flow caused by the pump malfunction under 'Inadequate Heat Removal' and 'Loss of flow through SFP HX' as shown in Table.

Table 1: Master Logic Diagram

| | | |
|---|---|---|
| Spent Fuel Pool Heatup | Inadequate Heat Removal | Loss of flow through SFP HX |
| | | Inadequate Heat transfer through HX |
| | | Inadequate Heat transfer to UHS |
| | Loss of Inventory | Loss from Cooling system |
| | | SFP Boundary failure |
| | | Seal failure |
| | Excessive Heat Load | |

.

### *2.2 Event Tree Analysis*

The Event tree analysis (ETA) is an analysis technique for identifying and evaluating the sequence of events in a potential accident scenario following the occurrence of an initiating event. By using the MLD, initial events are produced as shown in Table 1. One of the risk triplets, accident scenarios is modeled by constructing the event as shown in Fig. 3. The tree diagram consists of initiating events, mitigation systems, and relevant operator actions. There are two main success criteria in this study; 'Cooling Recovery' and 'Inventory Makeup'. This issue will be discussed in the next section.

### *2.3 Fault Tree Analysis*

The headings of Event tree is decomposed by modeling the fault trees consisting of the mitigating systems and the flag gates. The reference plant is located in the outside area of Pusan, Korea. The plant is under operation. There are alternate cooling systems, such as spent fuel pool cleansing system in the reference plant uniquely. If the inventory of spent fuel pool is lost, the drain of the coolant from the refueling water storage tank (RWST), condensate storage tank (CST) and demineralized water storage tank (DWST) might be available. There might not be mechanical failures events. The power recovery in certain time in the loss of offsite power accident sequence is considered for the minimal cut set quantification. The structural damage probability is also modeled in the external event scenario. The human error events are considered in fault trees as shown in Fig. 2. One of the fault trees used in this study is shown for the illustration purposes.

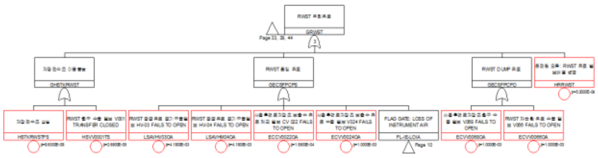

Fig. 2. Fault trees used for the RWST drain and dump line.

### *2.4 Human Reliability Analysis*

Human Reliability Analysis is important in this study. Most of the mitigating action needs to be performed by the operator manually. There are many handbooks such as NUREG/CR-1278 (THERP) and NUREG/CR-4772 (ASEP) to analyze the human error rates. The various operator actions bas been assessed like the Table 2 below. The THERP methodology has been applied to quantify the operator actions in this study.

Table 2: Quantified human reliability

| BE ID | Description | Prob. |
|---|---|---|
| HRSFPCR | Operator fails to recover SFP cooling system | 5.00E-2 |
| HRALTC | Operator fails to operate SFP cleanse system | 5.00E-2 |
| HRRWST | Operator fails to align line up for RWST | 5.00E-3 |
| HRDWST | Operator fails to align line up for DWST | 5.00E-2 |
| HRCST | Operator fails to align line up for CST | 5.00E-2 |
| HRMISOL | Operator fails to isolate SFP cooling system | 7.50E-2 |
| HRCRL | Operator fails to recover cooling system late | 1.47E-1 |

### *2.5 Initial Events in Plant Operation States*

The reliability data are required for quantifying the risk of a facility. There are two important plant operation states (POS) associated with the initiating events. They are fuel extraction operation and refueling states. During the refueling mode, the channel between the reactor cavity and the fuel building is connected. The initial events and their frequencies used in this study are described in the table below.

Table 3: Initial Event Frequency

| IE | Description | Freq (Op) | Freq (Ref) | Source |
|---|---|---|---|---|
| LSFP | Loss of SFP pool cooling | 2.40E-2 | 2.80E-1 | AEOD |
| LOOP | Loos of Offsite Power | 2.41E-2 | - | KOREA |
| LINV | Loss of Inventory | 2.00E-3 | 2.00E-3 | AEOD |
| EQE | Earthquake | 2.76E-4 | | KORI |
| PLOCA | Primary LOCA During Refueling | - | 1.93E-4 | AEOD |

### ***3. Results***

A level 1 PSA code has been utilized for quantifying accident scenarios. The one top fault tree has been constructed and evaluated quantitatively. The frequency of the occurrence of the spent fuel pool boiling is evaluated to be a value of $2.6\times10^{-5}$/Y. The most dominant cut-set results in the LINV-005 sequence as shown in Fig. 3. When the LINV Initial event occurs, operators need to restore the pool water level. The LINV-005 denotes a sequence, which operators fail to setup the SFP cleansing system for the inventory makeup. The Table 4 and Fig. 3 represent important cut-sets and the LINV sequences, respectively.

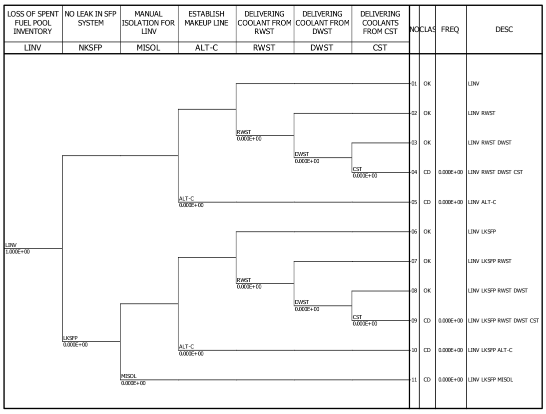

Fig. 3. Event tree for Loss of Inventory Initial event

Table 4: Dominant Cutsets

| Consequence | Basic Events | | | Frequency |
|---|---|---|---|---|
| LINV-005 | %LINV | HRALTC | | 1.00E-5 |
| LINV-011 | %LINV | HRMISOL | LKSFP | 3.00E-6 |
| LINV-010 | %LINV | HRALTC | LKSFP | 2.00E-5 |
| LOOP-004 | %TP | HRALTC | HRSFPCR | 6.02E-7 |
| LOOP-003 | %TP | HRDWST | HRSFPCR | 6.02E-7 |

## **4. Conclusions**

A methodology for assessing the risk associated with the spent fuel pool facility has been developed and is applied to the reference plant. It is shown that the methodology developed in this study might contribute to assessing these kinds of the SFP facilities. In this probabilistic risk analysis, the LINV Initial event results in the high frequent occurrence. The most dominant cut-sets include the human errors. The result of this analysis might contribute to identifying the weakness of the preventive and mitigating system in the SFP facility.

## **REFERENCES**

[1] N. Siu, S. Khericha, S. Conroy and S. Beck, H. Blackman, Lockheed-Martin Idaho Technologies Co., "Loss of Spent Fuel Pool Cooling PRA: Model and Results", 1996

[2] Swain A.D., U.S. Nuclear Regulation Commission, NUREG/CR-1278, "Handbook of Human Reliability Analysis with Emphasis on Nuclear Power Plant Application", 1983

[3] Jinkyu Han, KEPCO ENC, K34-PSA-INT-01, Internal Event Assessment rev2, 2008

[4] Beomsuck Kim, KEPCO ENC, K34-PSA-INT-05, Data Assessment rev2, 2008

[5] American Nuclear Society, "Fukushima Daiichi: ANS Committee report", 2012