A Eestimation of the Accident Sequence of the LOCA Groups for the PSA Model of the KSNP

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1. Introduction

A new trend of the probabilistic safety assessment (PSA) technology is to improve and enhance the current PSA model to be adequate for risk-informed applications (RIA). Requirements of a PSA model for the RIA are summarized as (1) reduction of the conservatism in the model utilizing all available information and (2) consideration of the specific features of a plant as designed, as operated [1]. This is because the PSA based on conservatism and insufficient consideration of the plant-specific features resulted in a shadow effect on the assessment results. When a PSA model is used in a risk-informed application, more precise risk-information is more helpful to decision making process, so the reduction of the conservatism and the consideration of the plant-specific features in a PSA model are the most essential elements. Recently, an effort has been performed to modify the current PSA model for the Korea Standard Nuclear Power plant (KSNP) to be used in risk-informed applications. A re-estimation of the accident sequence of the loss of coolant accident (LOCA) groups for the PSA model of the KSNP has been performed.

2. Four Major Re-estimation and Results

The LOCA sequences have been estimated as a part of the accident sequence of the KSNP. Four major parts of re-estimation are identified (1) the classification of LOCA groups, (2) recirculation of the safety injection, (3) auxiliary feedwater source, and (4) impact due to containment failure.

2.1 Classification of LOCA Groups

The first is the reclassification of LOCA groups. An accident sequence classification of the LOCA groups for the KSNP has been performed to obtain a suitable model for the RIAs. Because the current PSA model [2] for the KSNP was basically developed during the design phase of the KSNP, the plant-specific features were insufficiently incorporated in the PSA model. The accident sequence of the LOCA in the current model has been classified into three groups as (1) large break (above 0.2ft^2), (2) medium break (0.02ft^2 through 0.2ft^2), and (3) small break (below 0.02ft^2) LOCA. The result of this classification basically originated from a repercussion of the System 80 designed by combustion engineering, which is the prototype of the KSNP. In this paper, through the reviews of the final safety analysis report (FSAR) [3] and additional thermal hydraulic analyses by the MARS2.1 code [4], the frame features have been abstracted for the reclassification of the LOCA groups of the KSNP. The LOCA groups in the PSA study should be classified as considering three major safety functions: (1) reactor trip, (2) availability of the secondary heat removal, and (3) availability of the safety injection system. Table 1 shows a classification of the LOCA groups by the three major safety functions in a PSA model. Using the three safety functions criteria, the present study has changed the effective break sizes 0.2ft^2 to 0.5ft^2 for the large and the medium LOCA and 0.02ft^2 to 0.05ft^2 for the medium and the small LOCA, respectively. The reclassified LOCA sequences are accordance with each thermal hydraulic behavior shown in FSAR and the reference 6.

Table 1. A classification of the LOCA groups by the three major safety functions in a PSA modeling

LOCA groups	Reactor Trip	Safety Injection Systems	2ndary Heat Removal	Remark
Large	N/A	SIT, LPSI	N/A	
Medium	N/A	HPSI	N/A	
Small	A	HPSI	Applicable	

2.2 Recirculation of Safety Injection

The second is a change of the water source for the operation of the safety injection in the recirculation mode. When LOCA occurs, the safety injection systems operated automatically. The water source of the safety injection is the refueling water storage tank (RWT) including at least 650,000 gallons of water. If the water level of the RWT reaches at 7.5% of the full level, the engineered safety features actuation system (ESFAS) automatically rearranges the safety injection operation using water from RWT to re-circulate water gathered in the reactor sump. Available time for the safety injection to re-circulate operation was re-estimated by the FSAR and an available reference [6]. There is no different in case of the large LOCA, but it estimated as 3 to 5 hours in the medium LOCA and it estimated as 9 to 10 hours in the small break LOCA. Based on these reviews, the event trees for each LOCA groups have been reconstructed to reflect the new estimations.

2.3 Auxiliary Feedwater Source

The third is the arrangement of the water source of the auxiliary feedwater from the condensed-water storage tank (CST) to the other water source like a demineralized water storage tank (DWST) if the water in the CST has drained. The available time for this was estimated as 12 hours in the current PSA, but it in the emergency operation procedure (EOP) for the KSNP is indicated as 24 hours [5]. Based on this review, the event tree heading for the maintenance of secondary heat removal has been modified.

2.4 Impact due to Containment Failure

The last is the consideration of a core damage impact due to containment catastrophic failure. If the energy in containment discharges the environment at a LOCA, the containment fails due to a containment pressure rise. A heat sink through out the containment firstly considers the shutdown cooling system (SDS). If the SDS is unavailable, we use an indirect cooling by the containment spray system. If both systems fail, the containment pressure rises up to the failure pressure because any final heat sinks are unavailable. However, a time to containment failure is too long to consider that in the level 1 PSA because the acceptable primary mission time in an assessment of the level 1 PSA applies 24 hours. According to the MAAP3.0B code simulation, the containment failure time for the above mentioned sequence estimated as about 240,000 seconds [7]. It is noted that if the different mission times are considered at the same time for a PSA components and basic events have different importance. For example, estimations of the running failure with a different mission time have a different failure rate for each mission time. This makes a week point for consistency, so a core damage impact due to containment catastrophic failure is excluded in the present study.

2.5 Results of Re-estimation

Using the four major re-estimations, each event trees for LOCA sequences was re-constructed. A preliminary estimation of the core damage frequency (CDF) for each event trees has been performed. The results are shown in table 2. The present estimation doesn't have a detailed-information like a minimum cutsets but we obtained a reduced CDF from LOCA sequences by the new estimation.

Table 2. A preliminary estimation results of the core damage frequency of the LOCA sequences

	New Model* (/Ry)	The current Model (/Ry)	Remark
LLOCA	8.312E-07	1.05E-06	
MLOCA	8.219E-07	6.33E-07	
SLOCA	1.270E-07	1.86E-06	
CDF due to LOCA	1.780E-06	3.54E-06	

* The new model is only estimated by quantity of the point estimated heading values.

3. Concluding Remark

We show that the present re-estimation has a potentiality of the realistic estimation and a necessity of it using the recent information of the researches and the experience of the plant operation. However, to obtain a final precise PSA model to be used in the RIA, we should try to estimate the remaining accident sequences using a best-estimation code and to make a precise modeling including re-construct fault trees and cutest level estimation.

ACKNOWLEDGEMENT

This study was performed as a part of the '*long term research and development plan for the nuclear energy*' to be supported by the Ministry of Science and Technology, Republic of Korea.

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