Safety Effect Analysis of the Large-Scale Design Changes in a Nuclear Power Plant

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1. Introduction

Recently, Korea Hydro & Nuclear Power Company (KHNP) prepared for two units of nuclear power plants (NPPs) for long-term operation. For these units, large-scale design changes and system improvements were implemented even though the NPPs used old designs because KHNP focused on reinforcing the plant safety during their operation periods. These activities were predominantly focused on replacing obsolete systems with new systems, and these efforts were not only to prolong the plant life, but also to guarantee the safe operation of the units. This review demonstrates the safety effect evaluation using the probabilistic safety assessment (PSA) of the design changes, system improvements, and Fukushima accident action items for Kori unit 1 (K1).

2. Analyses using PSA Models

The analyses evaluated the seven design changes that were related to the plant specific PSA model from a total of 18 system improvements for safety enhancements and Fukushima action items. The results are explained through confirming the changes in the risk measures such as the core damage frequency (CDF) and containment failure frequency (CFF).

2.1. EDG Replacement

Emergency diesel generators (EDGs) are designed to power the safety buses in order to provide safety components with their driving power. EDGs are initiated at a loss of voltage (LOV) signal, and they are connected to the safety bus in order to provide power for the engineering safety features (ESFs) through opening and closing their load circuit breakers. K1 replaced both EDGs with a new EDG that had the latest technical requirements applied.

However, the failure experiences and operation time were insufficient to estimate the reliability of the new EDG. Therefore, the data in NUREG/CR-6928 [1] were used as the failure rate of the improved EDG in order to evaluate the CDF after the system improvement. For example, “Fail to Load and Run” in reference [1] is a failure mode in which EDGs are initiated and operated for less than one hour before failure. This failure mode was included in the “Fail to Start” mode of the PSA model in order to calculate the failure rate of the EDG. Table 1 includes the reliability data used in this evaluation. The changes in the system unavailability and CDF before and after the EDG replacement are reported in Table 2.

Table 1. Reliability data of EDGs

Component	Before Design Change		After Design Change		Failure Mode	Mission Time
	Failure Rate	Failure Probability	Failure Rate	Failure Probability		
EDG	1.15E-2/d	1.15E-2	5.00E-3/d	5.00E-3	Fail to start	-
	-	-	2.90E-3/d	2.90E-3	Fail to load & run	-
	1.59E-3/h	3.81E-2	8.00E-4/h	1.92E-2	Fail to run	24 h

Table 2. PSA effects of the EDG replacement

Risk Measure	Before Design Change	After Design Change	Change (%)
Unavailability of EDG	5.32E-2	3.05E-2	–42.07%
CDF	1.62E-05/RY	1.53E-5/RY	–5.56%

The review of the PSA effects of the EDG design change demonstrated that the CDFs were reduced in several initiating events including a loss of offsite power (LOOP) and station black out (SBO). The effect of the loss of power for safety components was reduced because the failure rates of the affected basic events were lowered through adopting the recent generic failure data in consideration of the EDG design improvements.

2.2. Movable Generator

A movable generator is a trailer-mounted generator that is equipped with a gas turbine. Movable generators are designed to be able to provide power to the emergency core cooling system (ECCS) as an alternative power source when a loss of emergency power sources such as EDGs and alternate alternating current diesel generators (AAC DGs) occurs under the LOOP condition. For the operation of this movable generator, a new termination box, which is powered by the movable generator, was installed in the EDG “B” room. In addition, a power cable was routed from the termination box to the terminating point for connection to the AAC DG. A new fault tree that considers the movable generator was added to the original PSA model.

For this design change, the fault tree was modeled by considering the cable path through which the movable generator power was connected to the AAC DG termination from the EDG “B” room. This fault tree was inserted into the failure logic of the power not being provided to the safety bus during a LOOP event because both EDG A/B and AAC DG fail simultaneously. In addition, this evaluation used the results of reference [2], which modeled an operator action in which the operator guided the movable generator and manually connected its cable to the EDG room. This operator action failure was analyzed by K-HRA [3].

Table 3. Reliability data of a movable generator

Components or Human Errors	Failure Rate	Failure Probability	Failure Mode	Mission Time
GTG	2.50E-2/d	2.50E-2	Fail to start	-
	1.00E-5/h	2.40E-4	Fail to run	24 h
Cabling	-	4.65E-1	Fail to connect GTG	2 h

The CDF comparison before and after the design change and operation improvements in consideration of the movable generator is as follows.

Table 4. PSA effects of the movable generator operation

Risk Measure	Before Design Change	After Design Change	Change (%)
Unavailability of Emergency Power	9.97E-05	6.83E-05	–31.45%
CDF	1.62E-05/RY	1.57E-05/RY	–3.09%

The PSA effect with regard to each initiating event was reviewed. It was found that approximately 3% of the CDF was reduced in the SBO event. This resulted from the safety improvements through preparing countermeasures for SBOs using a movable generator.

2.3. PAR Installation

The passive autocatalytic recombiner (PAR) of K1 removes hydrogen during design basis accidents (DBAs) and severe accidents. The PAR maintains the hydrogen density in the containment below 5% in order to prevent containment damages due to hydrogen deflagration.

The PAR was installed as a part of the system design improvements for ensuring the containment integrity during accidents. Thirty-four PAR units, which had different capacities according to their location, were installed inside the containment area.

The containment failure frequency (CFF) was recalculated through changing the fraction probability at the hydrogen burning point (head) on the decomposition event tree (DET) in the Level 2 PSA model. PAR does not have support systems such as electricity, instrumentation, or component cooling because it is a passive system. It was considered that PAR could control only late burning of hydrogen when all 34 PAR units are operated normally.

Table 5. Reliability data of PAR

Components or System	Failure Rate	Failure Probability	Failure Mode	Mission Time
PAR 1 Unit	1.00E-4/h	2.40E-3	Fail to operate	24 h
PAR System (34 Units)	-	8.16E-2	Fail to operate	24 h

The change in the CFF before and after the PAR installation indicated that the CFF decreased by approximately 0.69% and the late containment failure (LCF) due to hydrogen burning was reduced by 0.8%.

2.4. Design Change of Auxiliary Feedwater System

The auxiliary feedwater system (AFWS) supplies feedwater to the steam generator (SG) in order to remove the decay heat in the core during the emergency operation of a plant.

The design change of the K1 AFWS included a change in the location of the manual valves, which cause the turbine driven AFW pump to be unavailable during the system test due to their location; a change in the operating method for the AFW pump discharge valve, which changed from the automatic opening of the motor-operated valve (MOV) to the manual control in fully opened conditions; and the installation of cross tie pipes and valves between both trains in case of emergency. The CDF after this design change was reduced by 3.71% when compared with the value before the design change.

2.5. Improvement in the Main Steam PORVs

The power operated relief valves (PORVs) on the main steam line removes heat from the steam generator when steam relief to the condenser is impossible. The manual PORV was replaced with an improved PORV with a MOV design in order to assist operators to control the valves during emergency situations. Although the system configuration did not change, the reliability data demonstrated that the failure rate of MOV was six times higher than that of the manual valve based on the K1 PSA report. The evaluation results indicated that the CDF after this design change increased by 0.02%.

Table 6. Reliability data of a main steam PORV

Components	Failure Rate	Failure Probability	Failure Mode	Mission Time
Manual Valve	2.27E-8/h	1.80E-4	Transfer closed	-
MOV	1.40E-7/h	1.11E-3	Fail to remain open	-

Table 7. PSA effects of PORV improvement

Risk Measure	Before Design Change	After Design Change	Change (%)
Unavailability of System	5.57E-3	6.53E-3	+17.26%
CDF	1.616465E-05/RY	1.616734E-05/RY	+0.02%

3. Conclusions

For the large scale of system design changes for K1, the safety effects from the PSA perspective were reviewed using the risk quantification results before and after the system improvements. This evaluation considered the seven significant design changes including the replacement of the control building air conditioning system and the performance improvement of the containment sump using a new filtering system as well as above five system design changes. The analysis results demonstrated that the CDF was reduced by 12% overall from 1.62E-5/y to 1.43E-5/y. The CDF reduction was larger in the transient group than in the loss of coolant accident (LOCA) group. In conclusion, the analysis using the K1 PSA model supports that the plant safety has been appropriately maintained after the large-scale design changes in consideration of the changed operation factors and failure modes due to the system improvements.

References

[1] NUREG/CR-6928, Industry-Average Performance for Components and IEs at U.S. Commercial NPPs, NRC, 2007.

[2] Analysis on safety improvement effects on Fukushima action items, KHNP, 2012.

[3] TR-2961, Development of Standard HRA Method (K-HRA) for NPPs, KAERI, 2005.