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The Features of Probabilistic Safety Assessment in Standard Design of Korean Next Generation Reactor

Abstract

As the usefulness of the Probabilistic Safety Assessment(PSA) has been proved during the Individual Plant Examination(IPE) implementation and other PSAs, it has been recommended to use the PSA as a design tool to optimize the plant design in terms of the safety. And, there are several characteristics of KNGR PSA which is different from those of existing plants in technical contents, analysis scope or its purpose. This paper presents some experiences of KNGR specific PSA interaction in view of on-going designs such as periodic system analyses and their design feedback, human reliability analysis(HRA) and human factor engineering(HFE) for advanced control room and man-machine interface design, and severe accident mitigation design and probabilistic assessment to confirm the technical feasibility for assuring the accident mitigation prioritization of system. Also, it explains design specific analysis results of external event and design reliability assurance program(D-RAP) concept to assure the safety of PSA that has been applied at the stage of the basic design of the KNGR. This paper gives some example and guidance for the future PSA to be performed in relation to vigorous risk informed application (RIA).

I. Introduction

The PSA has usually been performed for operating plants. Even in cases of PSA for new plants, the assessment is performed to confirm the safety of the plants after the design is completed. Therefore it is not easy to change the design based on the findings from PSA. In other words, the roles of PSA for new plants is limited to supplementing the deterministic design evaluation. To use PSA as a design tool, the assessment is required to start at the early phase of the design based on assumptions and judgments of the analysts due to the lack of the plant specific information.

KNGR PSA has been started to assess and enhance the safety of on-going design as a family of integrated plant level analysis in combination with economics, constructibility, and performance analysis. And, there are several characteristics of KNGR PSA different from those of existing plants in technical contents, analysis boundary or its purpose. The design and analysis are performed concurrently and information on them are exchanged to improve the safety of plant. The PSA results are included in standard safety analysis report (SSAR) as a licensing requirement. Also, there are several specific spheres to be discussed such as HRA/HFE interface of PSA and MMI design, reliability assurance program, containment performance analysis in consideration of deterministic severe accident mitigation design as well as traditional existing system analysis and its interaction with design modification. Table 1 shows the characteristics of KNGR PSA and its application compared with those of the conventional plant.

Table 1. Comparison of standard design and PSA interface with that of conventional plant

Items	KNGR	Conventional Plant
Analysis schedule for Licensing	- PSA results are incorporated into SSAR Ch.19 with severe accident mitigation design - Concurrent submittal of PSA with design document	- Separated PSA report to SSAR - Preliminary PSA report after 12 month of PSAR - Final PSA report after 6 Month FSAR
Scope of Analyses	- Design specific PSA - Full scope Level 3 PSA - Full/shutdown mode included	- Site specific PSA - IPE or Level 2 PSA required - Shutdown PSA for new plant
Analysis Methodology	- HRA/HFE interface into MMI - External Event Fire PSA based on FIVE PSA based SMA Flood PSA	- HRA for PSA quantification - External Event Fire PSA Seismic PSA Flood PSA
Design/Operational Safety Assurance	- Reliability Assurance Program at design certification stage	- Maintenance Rule, Periodic Safety Review at operational stage

II. Examples of Interaction Between PSA and Design

II.1 Design Specific System Analysis for Standard Design

During the KNGR standard design, the PSA was performed mainly to confirm the safety goal established by Korean utility requirements document and also to identify any design weaknesses. For this purpose, periodic PSA was performed during the three phases of the design.

The experiences of past PSAs, especially those of Ulchin units 3&4 and System 80+ were used to identify the vulnerabilities in previous plants. Examples are the Passive Secondary Condensing System(PSCS) to reduce the risk from loss of auxiliary feedwater and double containment to increase the radiological performance and public confidence. These feasibility studies for new design were done using the PSA model of the reference plant during the conceptual design phase.

The first assessment using KNGR model at basic design was done in June 1997 when the design was in the middle of the second phase. Design information for the KNGR evaluation was obtained from the System 80+ and Ulchin Units 3&4 and the analysis is made to the PSA Level 1, core damage frequency. Although the core damage frequency at this assessment was much lower than that of the safety goal, some design improvement items were identified through intensive review of the results. This information was handed to the design teams and was adopted for design improvement.

In design optimization study at end of basic design, preliminary evaluation was performed to identify the impact of PSCS and double containment removal and its alternative designs. Even if it was not easy, it was determined to remove these safety significant systems for advanced design concept of plants to enhance the economic viability compared with its competing energy sources. Alternatively, some design options having economic competitiveness are derived. These are the non-safety graded diesel driven pump and external reactor vessel cooling(ERVC) strategy. In addition to the big change from PSCS and double containment design to diesel driven pump and single containment respectively, some trivial design changes such as separation of control power to auxiliary feedwater system but affecting somewhat large on safety was reviewed.

The analysis results were documented and summarized so that it can be used by the decision making process of each design phase. The detailed result of the PSA of the basic design has been presented at the PSA 99. The examples of PSA interaction of design process in full power internal event analyses are showed in Table 2. These integrated safety evaluation results and design changes from phase 2 KNGR PSA are transferred to the betterment of Korean standard nuclear power plant plus(KSNP+). This is possible from the fact that the safety is greatly improved and the cost of design improvement for KSNP+ is not too high.

The results of the containment performance analyses in terms of the conditional probability of containment failure and source terms indicate that KNGR design does not have any particular vulnerability to containment performances compared with other PWR plants. The effectiveness of new design features of KNGR containment system such as advanced design of cavity configuration and cavity flooding system(CFS), hydrogen mitigation system(HMS), external reactor vessel cooling system(ERVC) and emergency containment spray backup system(ECSBS) are modeled in containment performance analysis.

The characteristics of KNGR design against containment failure frequency are as follows; The KNGR cavity configuration allowed much less corium ejection out of cavity during high pressure melt ejection compared with conventional PWR. Hydrogen igniters and passive auto-catalytic re-combiners could prevent hydrogen burn. Also, the CFS was very effective to mitigate corium-concrete interaction(CCI) and basemat melt-through, whereas the ECSBS plays ultimate heat sink of containment at late phase of accident. In the current KNGR containment design, there is a provision that the coolant from IRWST can flow into the cavity and provide external reactor vessel cooling. According to sensitivity studies, this provision is somewhat effective to the containment performance even though the passive cavity flooding system and outer containment are removed in design optimization process. The assessment results showed the KNGR satisfied the safety goal in view of plant protection and severe accident mitigation.

Table 2. Design and PSA interaction in each phase of KNGR design status

Design Stages	Design process	Interaction of PSA
Conceptual Design (1992.12 ~ 1995.2)	- Advanced design features · 4 Train SIS - Passive Design Features · PSCS, Fusible plug in CFS · FD in SIT, PAR	- Confirm Safety Goal by K-URD - Feasibility study of ADF/PDF are performed
Basic Design (1995.3 ~ 1999.2)	- AFWS check valve removed - SDS/PSV to POSRV - Capacity of PSCS changed	- Design evaluation for each design changes
Feasibility Study for Optimization (1999.1 ~ 1999.8)	- PSCS removed - Double containment to Single Containment	- Alternative design options are reviewed - Design leak rate and Exclusion area boundary is adjusted - ERVC(External Reactor Vessel Cooling) adopted
Design Optimization (1999.3 ~ 2001.12)	- AFWS divisional cross-tie - CCWS divisional cross-tie	- Safety impacts of design change and operational effects are reviewed

II.2 Human Reliability Analysis for Advanced Control Room

There are two purposes in human reliability analysis(HRA) performed during KNGR design. One is the quantification of human error probability for PSA and the other is to provide critical operator actions for human factor engineering(HFE). The latter purpose is needed to meet a new design requirement for advanced control room according to NUREG-0711. Thus HRA, a part of PSA was incorporated into KNGR HFE design.

Critical operator actions are identified based on importance analyses usually used to identify the items representing the significant change in plant safety. For KNGR HRA/HFE, RAW(Risk Achievement Worth) grater than 2.0 or RRW(Risk Reduction Worth) grater than 1.05 derived from PSA results were used to define critical operator actions.

The critical operator actions listed in Table 3 are based on basic design model. They are addressed in the design of plant MMI, procedure development, and training in order to minimize the likelihood of personnel error and to provide for error detection and recovery capability. As described in II.1, the KNGR system has been significantly changed through the optimization process. Currently, phase III of the design is underway and the PSA is being performed using the updated design, so the critical operator actions may be revised and will be incorporated into HFE design evaluation.

Table 3. Critical operator actions for MMI design

Operator Actions	RRW	RAW
Operator fails to perform ASC during small LOCA(HR-ASC-SLOCA)	1.183	1.98
Operator fails to initiate emergency boration using charging pump(CVH)	1.139	6.66
Operator fails to perform POSRV operation early(SDE)	1.139	6.66
Operator fails to initiate Hot Leg Injection (HR-HLI)	1.065	220.9
Operator fails to perform ASC during SGTR(HR-ASC-SGTR)	1.051	2.65
Operator fails to maintain secondary heat removal (HR-MSHR)	1.014	7.60
Operator fails to align SCS for long term cooling(SCOPH-LTC)	1.014	3.99
Operator fails to align SCS for injection(SCOPH-INJ)	1.013	3.82
Operator fails to perform Pressure Control(PCL)	1.012	83.77
Operator fails to close SG2 MSADV at late phase(ADL)	1.012	83.77
Operator fails to perform 4 POSRV operation early	1.003	2.06

II.3 External Event Analyses for Standard Design

The external events analyses for the KNGR design were performed. Bounding site characteristics were used to minimize potential future restrictions on plant siting. Because sufficient plant design information such as equipment lists, cable layouts, fire detectors types, location of penetrations was not available at standard design stage, the plant risk from fires was estimated from some conservative assumptions. Especially, the cable routing was one of the important information to perform the fire risk analysis which was not available at this design phase. To solve this problem, power and control cable routes were assumed with support of electrical and I&C designers based on electrical load list, the selected equipment location on plant general arrangement drawing and piping plan drawings with design basis philosophy for separation of the components and cabling associated with safety and non-safety related systems.

The internal fire risk analysis at power operation shows that the design philosophies for the KNGR system, plant layout and separation of redundant equipment resulted in a low fire-induced core damage frequency(CDF) compared with that of existing plants. The major reasons for low fire-induced CDF, despite the conservative assumptions and engineering judgements due to lack of detailed plant design information, include the following:

- The fire protection design provides separation of the redundant safety-related components and cables using 3-hour rate fire barriers. For example, areas containing safety-related cables or components are physically separated from one another and from the areas that do not contain any safety-related components by 3-hour fire barriers. This feature diminishes the probability of a fire to impact more than one channel of safety-related system.

- Unlike main control room(MCR) of existing plants, the contribution of KNGR MCR to the overall fire-induced CDF is estimated significantly low because MCR is designed to provide redundancy in MCR operation; that is, if the operator evacuation from MCR is not required, the alternative means is still available to shutdown and control the plant within the MCR.

The approach applied to the KNGR for seismic risk assessment is the PSA-based Seismic Margin Assessment(SMA). The basic difference between a PSA-based SMA and a seismic PSA is that the SMA perform no seismic hazard analysis and thus does not produce any core damage frequency, while the PSA approach produces this value. The NRC methodology was adopted for KNGR analysis. It is due to the fact that the EPRI SMA based on success criteria does not produce sufficient insight to be used as a design tool. And the seismic PSA is known to be governed generally by the seismic hazard which has a lot of uncertainties and thus the results of the assessment could be misinterpreted. The PSA-based SMA is used to provide insight that can be obtainable from a seismic PSA without involving of large uncertainty contained in the seismic hazard analysis.

II.4 Reliability Assurance Program for KNGR

Reliability assurance program(RAP) is applied for KNGR to ensure that the safety and availability evaluated in the PSA and availability study at the early phase of the design are maintained throughout the plant life cycle. The program is based on risk-based design, procurement, construction, and operation for important system, structure and components. So, it is related to the RIA(risk informed application) such as risk informed in-service inspection(RI-ISI), RI-Tech. Spec. change, Graded Quality Assurance(GQA), etc. It focuses on enhancing the competitiveness of nuclear industry against other electricity generating facilities. The difference from other program such as PSR(periodic safety review) and maintenance rule(MR) is that it focuses the safety and unavailability of SSCs(System, Structure, and Components) from the initial design stage.

The concept and philosophy of RAP are based on standard design of one step design certification process and incorporation of PSA results and their risk information into existing deterministic design and decision process. Due to pendency of the plant design at the time of the initial concept, there may be concern about the validity of results from either a preliminary PSA or RAM model. As the design information about the plant becomes better defined, the model will be improved. As the final plant design is approved for construction, these analytical tools can be used to answer questions raised during the licensing process and to resolve plant design changes on a rational basis.

As the plant commences its operating life, it will begin to show where vulnerabilities may lie and design changes can be made to prevent their re-occurrence. The information initially available to serve O-RAP becomes increasingly concrete as plant operating experience accumulates. As a result, O-RAP requires a change in perspective from reliability assurance during design and these differences result in need for a different approach towards plant reliability assurance during each phase of its life. At this time, based on the KNGR PSA model of basic design, the SSCs are selected for use in the detailed design. After the review and comment are resolved from engineering discipline, the final SSCs for the RAP will be distributed to all designers for use in the detailed design and O-RAP related activities.

III. Conclusions

The KNGR PSA has been performed based on standard design which is somewhat different scheme compared with conventional plants. The merit and shortcoming of KNGR PSA has been explained with some specific examples. The KNGR specific PSA features discussed in this paper are HRA/HFE interface of PSA and MMI design, reliability assurance program for design stage, containment performance analysis in consideration of deterministic severe accident mitigation design in addition to the traditional existing system analyses and their interface with design modification.

The external events analyses for the KNGR were performed using bounding site characteristics. The preliminary results show a greatly increased safety compared to the conventional plants by virtue of design characteristics such as divisional area and safety/non-safety equipment rooms separation and increased seismic capacity of 0.3g. But, the detailed safety of external event for standard design will be evaluated when the detailed design information such as construction and site specific data are secured.

The HRA activities in KNGR PSA and HFE design were reviewed and critical operator actions were identified based on importance analysis. The HRA-dominated sequences in KNGR PSA were intensively considered in MMI design.

RAP in KNGR confirms that the predicted reliability for a specific plant design will meet or exceed each of the performance goals prescribed in the utility requirements document. And it also guides the design optimization process and ensure that all resources expended in making design improvements in safety, reliability or performance provide a positive return. The approach presented here is a preliminary one. Needed is an additional study which includes potential problems in applying the program to actual project and interfaces with other issues such as graded QA, interactions with other design organization to make it final.

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