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Insights from the KNGR Preliminary Level 1 Probabilistic Safety Assessment

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Abstract

Korean Next Generation Reactor(KNGR) is a standardized evolutionary Advanced Light Water Reactor design under development by Korea Electric Power Company (KEPCO). It incorporates design enhancements such as active and passive advanced design features(ADFs) to increase the plant safety. A preliminary level 1 Probabilistic Safety Assessment(PSA) has been performed for the KNGR to examine the effect of these safety features. The preliminary PSA result shows that it meets the KNGR safety goal on core damage frequency(CDF). The result of this safety assessment shows that the four-train safety systems, and the ADFs such as Passive Secondary Cooling System (PSCS) contributes greatly to the reduction of the CDF. Furthermore, several design changes are made or proposed for detailed review based on the PSA insights.

Ⅰ. Introduction

To cope with the 21th century national energy demand, KEPCO launched the next generation reactor project in 1992 to develop the standard design of advanced pressurized water reactor by 2001. Enhancing safety and economics are considered as the continuing goals in this project. The safety objective is that a newly designed plant will not noticeably increase existing public risk. PSA techniques are being employed for evaluating the safety of the plant and optimizing design features.

Considering the risk to the general public, the PSA goals are set to be less than 1.0 x 10^{-5}/RY for the core damage frequency (CDF) and to be less than 1.0x10^{-6}/RY for the large release. The large release stands for the radioactive exposure greater than 1 rem for 24 hours at the site boundary. Table 1 lists contents and design goal of safety which is a part of top-tier requirements of Korean utility requirement document(URD).

Table 1 Safety Requirements of KNGRs

Items	Design goal	Remarks
· Core damage frequency(CDF)	· Less than 10^{-5}/RY	· High pressure or Single accident scenario CDF less than 10^{-6}/RY
· Containment failure frequency	· Less than 10^{-6}/RY	
· Radiological release limit target	· 1 rem/24Hr at site boundary, and less than 10^{-6}/RY	
· External events	· Seismic, Flood, Fire	
· Shutdown/Low power PSA	· Simplified analysis	· Seismic excluded in quantitative goals

Prior to 1995, probabilistic safety scoping analysis were performed on the KNGR conceptual design. In that analysis, the feasibility study of possible incorporation of passive design features(PDFs) into KNGR was performed. Four passive design features as well as the other advanced active systems were considered as candidates. One key features of this project is to examine the safety, economy, and operation & maintenance aspect during the design stage. In basic design phase, the PSA is scheduled to be performed three times to periodically evaluate the design features and to feedback the result to the design. The first evaluation was performed in 1996. Even if its scope is very limited, it showed some important design areas of KNGR where the design effort should be focused. The second phase of safety assessment are performed by cooperative effort between Korea Electric Power Research Institute (KEPRI) and Korea Power Engineering Company (KOPEC) which is designing KNGR NSSS and BOP systems. This paper introduces the result of the second evaluation. Especially, we focus on the role of PSA in the design process.

II. Description of ADF/PDFs for PSA

The philosophy of enhancing the safety is based upon three levels of safety. The first level of safety comes from designing a system with a high degree of reliability which has a large safety margin. The second level of safety is to provide means that will forestall or cope with abnormal events. The final level of safety is based on the safety features to mitigate a set of severe accidents even though they are considered highly unlikely. Some of the design features which fulfill these safety goals are summarized in Table 2.

The design temperature in the hot leg is reduced to 615°F from 621.2°F. Low coolant temperature will increase the overall operating margin. The increase of pressurizer(PZR) and steam generator(SG) water inventory will also smooth out the transients in case of the loss of heat sink.

In relation to the prevention and mitigation of severe accident, KNGR has adopted the design features, such as four-train safety injection system(SIS), in-containment refueling water system(IRWST), safety depressurization system (SDS), auxiliary feedwater water system(AFWS). These features were not used in existing nuclear power plants (NPPs). The four train SIS with direct vessel injection(DVI) are directly connected to reactor pressure vessel and supply the emergency core cooling water drawn from IRWST. The SIS design is to achieve higher reliability through simplification and diversification by merging high pressure, low pressure, and recirculation modes of operation into one single safety injection mode. The reliability of the containment spray system is enhanced by an exchangeability between shutdown cooling pumps and containment spray pumps, and by providing separate independent heat exchangers. The reliability of AFWS is substantially increased by designing two 100% motor-driven and two 100% turbine-driven pumps. Furthermore, two independent safety related auxiliary feedwater storage tanks are provided for its water supply in addition to existing condensate storage tank.

Based on the qualitative evaluation performed in the phase Ⅰ, the following four passive systems have been considered in the phase Ⅱ for design incorporation in KNGR; 1)Passive secondary condensing system(PSCS); 2)Fluidic devices in safety injection tanks(SITs); 3)Catalytic hydrogen igniters; 4)Passive cavity flooding system. Currently, the KNGR adopted the above active or passive safety features against severe accidents. Of these four systems, PSCS (Figure 1) which supplements AFWS is modelled in this analysis in detail. PSCS takes inlet flow from the steam line and returns condensate into the feedwater line after condensation through condenser tubes.

Table 2 KNGR Advanced Design Features Which Enhance the Plant Safety

Function	Design Features
Inherent Safety Margin	· Larger RCS Coolant Inventory(Reactor Vessel, Steam Generator, Pressurizer) · Hot Leg Temperature Reduction
Prevention of Core Damage	· 4 Train SIS/Direct Vessel Injection, IRWST · Improved AFWS with Tank Inside the Auxiliary Building · Residual Heat Removal Capability(PSCS, SCS/CSS) · Alternating Alternate Current(AAC) to Cope with Station Blackout
Mitigation of Severe Accident	· Prevention of Direct Containment Heating(DCH) + RCS Pressure Reduction by Safety Depressurization System + Prevention of Core Melt Progression to Containment at Failure of Reactor Vessel · Hydrogen Ignitor · Corium - Concrete Interaction(CCI) Reduction + Corium Coolability Increase by Reactor Cavity Floor Area Enlargement(0.02 m^2/MWth) + Cavity Flooding System · Double Containment with Annulus Ventilation System

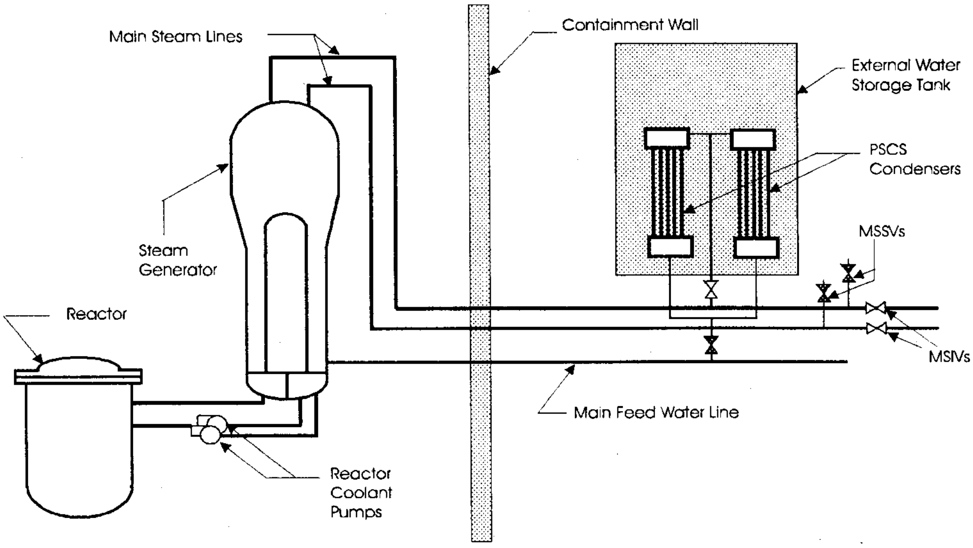

Figure 1 Schematics of Passive Secondary Condensing System

III. Analysis methodology

The PSA methodology applied in this study is based on the EPRI URD Volume II, Appendix A (EPRI URD PRA KAG). The major assumptions and approaches in performing this analysis are as follows:

1) The analysis is performed with the design as of June 30, 1997. This corresponds to the 40% completion of the design. To identify the discrepancies between the design and PSA modelling, interaction meetings were held with the designers. Based on the review comments from the designers, the PSA model was revised and requantified in October with the design level of 50%. The systems that has not been designed in detail are modeled with the help of the designers.

2) The initiating event category and its frequency are based on EPRI ALWR initiating events database. Since the total trip frequency goal for KNGR is 0.8/year, the frequency for the general transient is adjusted accordingly.

3) The failure data such as component, common cause failure(CCF), human reliability analysis(HRA) are based on EPRI URD PRA KAG and the results of System 80+ PSA.

4) The major objective in this interim analysis is to identify design improvement and design effect of KNGR on the safety goal. Hence, the focus is comparing the plant safety or system reliability with domestic standard plants or other advanced light water reactors. A detailed comparison on system level is performed by examining the system characteristics of KNGR against that of other ALWR and Korean Standard Nuclear Power Plant (KSNP). Specifically, the effect of four train SIS, IRWST, AFWS, PSCS, and CSS/SCS on CDF has been examined in detail.

Ⅳ. Results of the preliminary PSA

In this preliminary study, the CDF caused by internal events was estimated 7.9 x 10^{-7}/RY. As shown in Figure 2, the significant initiating event is medium LOCA(1.9 x 10^{-7}/RY, 24.7%). Other important initiating events are steam generator tube rupture (1.5 x 10^{-7}/RY, 18.6%), reactor vessel rupture (1.0 x 10^{-7}/RY, 12.7%), large LOCA (9.8 x 10^{-8}/RY, 12.5%), small LOCA (9.7 x 10^{-8}/RY, 12.4%), and SBO/LOOP((9.3 x 10^{-8}/RY, 11.87%). The key results obtained in this preliminary study are :

1) There is no dominant contribution to total CDF by single initiating event. The CDF by LOCA events including SGTR is evenly distributed, each event contributing about ~10^{-7}/RY except interfacing system LOCA whose value is negligible to ~10^{-10}/RY.

2) The transients including SBO/LOOP, loss of feedwater system contributes to 20% of CDF of KNGR. This is mainly due to the increase in heat removal capability by PSCS and other advanced design features such as dedicated shutdown cooling system and auxiliary feedwater system. It is estimated that the PSCS reduces the CDF by an order of magnitude for the transient initiating events.

3) By reducing the transient induced CDF, the relative CDF contribution by LOCA is somewhat increased. However, the absolute value has been reduced substantially due to the improved design of four-train SIS and IRWST.

4) The LOOP, including station blackout(SBO), is less of concerns in KNGR because it has very small contribution to the CDF. This is due to the addition of AAC and due to other design changes.

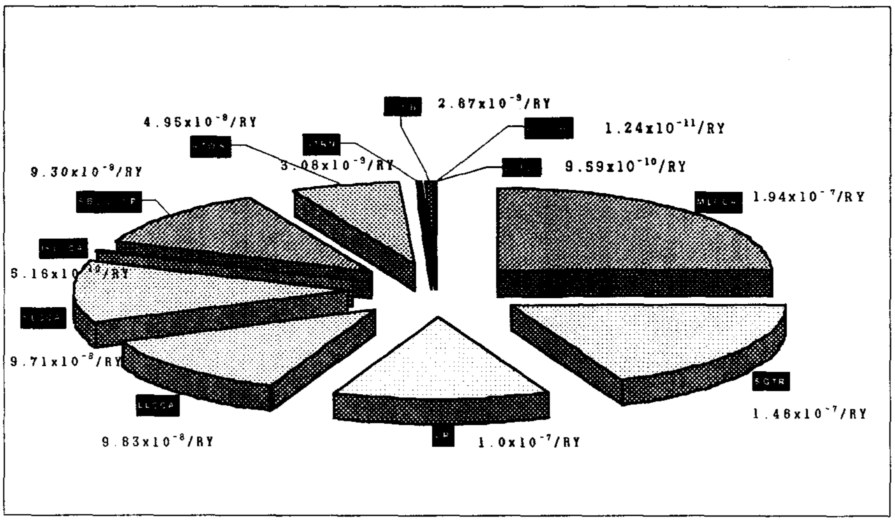

Figure 2 Core Damage Frequencies for Major Initiating Event of KNGR

When a single contributing events are examined, there are five events that contribute 10% or more to the CDF. These five events contribute up to 70% of CDF.

1) The largest one is medium LOCA with failure of hot leg injection after 2 or 3 hours of successful operation of DVI injection. The dominant contributors are a common-cause failure of hot leg isolation valve and operator fails to initiate operation.

2) The next important sequences are SGTR and small LOCA with failure of aggressive secondary cooling. These are also high pressure sequences that are important to the containment analysis. These sequences are due to the concurrent failure of safety injection system and secondary feedwater system. The dominant contributer in these scenarios is the common-cause failure of safety injection isolation valve and the failure of operator to initiate aggressive secondary cooling.

3) The next but important sequence is station blackout with failure of recover of offsite power. By comparison with the other ALWR PSA results, KNGR showed better results in CDF except this scenario. It is due to the combined effect of the differences in design and the modelling difference in maintenance, operating mode of diesel generator, and alternating AC.

To examine the system reliability of KNGR advanced design features, systemwise comparison is performed between KNGR and System 80+ as well as Ulchin unit 3&4. In cases when 1 of 4 direct vessel injection is required for small LOCA or SGTR, the unavailability of KNGR SIS is 4.12E-4 whereas that of Ulchin unit 3&4 is 3.32E-3. This decrease in KNGR is due to the redundancy of SIS trains and elimination of recirculation mode by IRWST. Unavailability of the auxiliary feedwater injection during loss of feedwater transient is examined. In this case, the unavailability of KNGR AFWS is 1.22E-4 whereas that of System 80+ is 5.84E-5. This difference is due to the common-cause failure of check valves in auxiliary feedwater lines. Based on this analysis, it is recommended that the designers eliminate the check valve of auxiliary feedwater lines.

Since the KNGR design is still underway, the PSA analysis is performed with several assumptions. To examine the effect of assumptions, a sensitivity study is performed. The sensitivity study considers the effect of major design changes such as PSCS, SIS, AAC as well as the effect of common-cause failure or HRA modelling. 1) Taken into account the accident being affected by PSCS, the PSCS reduces the CDF by about 67%. PSCS is very effective for the transients which require secondary cooling system. The most affected accident sequences by PSCS is loss of feedwater and general transients and station-blackout -induced transients. 2) In KNGR, alternating alternate current(AAC) is adopted to cope with the station blackout. Sensitivity study showed that the CDF doubles when the AAC is not credited. 3) When we review the PSA of existing plants and the System 80+, we find some discrepancies in CCF modelling and HRA data. The sensitivity analysis showed that CDF is greatly influenced by CCF and HRA modelling. In this study, the CDF if not considered the CCF is 2.77E-7/RY. And when it is increased the human error probability by a factor of 10, CDF increases to 3.9E6/RY. Careful consideration in system modelling is planned for further analysis.

V. Conclusions

The preliminary CDF for the KNGR internal events is 7.9E-7/RY. KNGR has adopted several design features such as quadrant separation or fire barriers that would reduce CDF for external events. The CDF for external events is expected to be low compared to that of the existing plants. Hence, the CDF for KNGR would be well within the ALWR goal of 1.0E-5/RY.

PSA has been an integral part of the KNGR design process. Several design changes were made or considered for further study as a result of this preliminary PSA results. Examples are: the elimination of check valves to increase the AFWS reliability, change of automatic operation logic in startup feedwater system, some addition of PSCS steam extraction or feed line isolation valves, and connection of containment spray backup system to more reliable position.

The effect of design change of electrical power supply system incorporating the auxiliary switchyard is evaluated. The analysis results show that the system is robust compared to the previous plant against electrical power loss transients. This is due to the additional AAC and design improvement of turbine generator system and DC power supply system. Even though, it is somewhat less reliable than that of System 80+, it is recommended not to consider any further design improvement in electrical power supply system in view of safety concerns.

Although passive system such as PSCS showed significant safety enhancement, the final conclusion has not yet reached. The performance and economic analysis need to be performed for the better designs.

Another important improvement of KNGR to safety is the HRA. KNGR will employ an advanced control room with digitalized I&C system and incorporation of man-machine interfaces. Continuing effort in designing control room with updated technology and development of proper procedure for operation and emergency preparedness is expected to reduce human errors.

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