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Availability Improvement Program and a Role for the Safety

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Abstract

The safety and availability are programmed in KNGR(Korean Next Generation Reactor) design from the conceptual design stage. The goal is envisioned as one of Top-tier utility requirements. To achieve the plant availability, the RAM (Reliability, Availability, and Maintainability) program was established. This program which, from the outset of the design phase, systematically assesses the forced outage and planned outage and provides insights to the design process to improve plant availability. This paper is to present the results from the RAM analysis for plant availability and their effects on the plant safety emphasizing its role for the safety enhancement during the next design phase.

Ⅰ. Introduction

The KNGR project is aimed at providing the standard design of an advanced pressurized water reactor (ALWR) by 2001. The development consists of three phases. During the first phase from 1992 to 1994, the selection of preliminary design concepts was made and the top-tier design requirements were developed. To ensure these top-tier design goals, the Korea Electric Power Corporation (KEPCO) has organized the plant level analysis group to view the top levels of design verification such as safety, economics, constructability, performance, and radiation protection of the plant in the basic design stage (phase II).

As one of the plant level analysis topics, the plant performance is assured in terms of power production capability (i.e. plant availability) through the RAM program. The design impacts on the plant forced outage and planned outage are assessed to maximize the plant availability, to predict if the top-tier goals of plant availability factor of 90% and trip frequency of 0.8 per year is achievable, and to enhance plant safety at power and shutdown operation modes.

Ⅱ. Review of the Availability Assessment for KNGR

1. Forced Outage Evaluation

1.1 Review of Operating Experiences for Korean Nuclear Power Plant

The forced outage data of the existing plants are essential factors to predict the forced outage aspect for the KNGR. For this, trip records are investigated for the domestic PWRs (Kori Unit 1,2,3,4, YGN Unit 1,2,3,4, and UCN Unit 1,2). Trip causes and failure modes, time to restore, and the countermeasures against reoccurrence are identified for each trip case occurred during the period from 1978 to 1996 based on the annual trip reports published by KEPCO. The trip cases occurred during the first one year of commercial operation were excluded in the analysis considering the bathtub curve effect to get insights of plant lifetime.

Table 1 shows system level plant trip frequency trends. Turbine/Generator and their auxiliaries, the most dominant contributor among the major 19 systems, caused plant trip as many as 71 times(32%) during the period followed by Feedwater System, Electrical Power System, CEDM, and Reactor Coolant System. These 5 systems take up the plant trips as much as 190 times (85%). Table 2 illustrates component level plant trip frequencies. The most contributive components, main turbine and generator, caused plant trip as many as 52 times(23%). The next most contributive components are Feedwater Control Valve, CEDM control circuits and so on. Total trip events resulted from the functional failures of top 10 components amounts 146 times (65%) during the period. Table 3 shows the trend and result of the root cause analysis. Component functional failure, external event, and human error account for 76.7%, 5.8%, and 14.8% respectively.

1.2 Forced Outage Evaluation

KNGR forced outage was preliminarily evaluated based on the operating experience study. The focus is to estimate the number of the plant trip by the same root cause of the each trip case. Since the KNGR is at the basic design phase, system designers, equipment vendors, and system engineers of the plants were consulted to predict detailed design and to confirm design modifications and equipment refurbishments in each plant. Some causes were judged not to lead to plant trip in the KNGR. For example, 5 trip cases occurred due to the failure of RTD (Resistance Temperature Detector) bypass line of Reactor Coolant System. In the KNGR as well as some of current plants, the thermowell type rather than RTD is adopted for measuring temperature of the coolant, whose cases will not occur in the KNGR. After screening the failure causes for the KNGR design, the preliminary estimation on trip frequency is made to be 0.8 per year and 2 days per year on forced outage duration.

2. Planned Outage Evaluation

Planned outages, during which refueling and maintenance (R/M) is performed, is the most dominant factor to the plant unavailability. The operating experience shows that the planned outage duration is characterized by the plant types and maintenance activity levels such as normal outage and extended outage. Therefore, a typical 45-day normal outage schedule for a 1,000 MWe PWR was chosen as a basis to assess the planned outage of the KNGR. The RAM analysis has established the normal outage duration of the KNGR as 42 days by reducing the reactor vessel head handling work loads with little impact to the maintenance activities of the other major components like Steam Generators. Figure 1 shows the normal outage schedule of the KNGR which will be further updated during the detailed design.

KNGR has simplified reactor vessel head area by utilizing an integrated reactor vessel head package. An Integrated Head Assembly (IHA) is designed to incorporate all of the reactor vessel head components into one module(Figure 1). The IHA casing is designed so that one can use the multiple stud tensioner during refueling outage. The multiple stud tensioner allows simultaneously detensioning and tensioning of all the reactor vessel studs. This enables the removal of the head area components and reactor vessel head at once. The use of IHA is estimated to save almost 3 days in comparison to the typical seven-day schedule of existing plants. Prior to refueling in existing PWRs, the equipment on reactor vessel area has to be removed and temporarily stored before removing the reactor vessel head. This process has resulted in increasing the overall refueling duration as well as the personal radiation exposure. An IHA is being designed to consolidate the following into an one-package component design: the head lifting rig; lift columns; missile shield; CEDM forced air cooling system; electrical and instrumentation cabling; insulation and reactor vessel head. The IHA lifts the reactor vessel closure head and the head area equipment at one time. Therefore, the amount of critical path time required to reach the reactor vessel internals can be reduced.

Furthermore, the KNGR will have advanced design features such as a permanent pool seal (PPS) and a quick opening fuel transfer tube blind flange (QOBF) to reduce refueling work load. The PPS is installed between the reactor pressure vessel and the surrounding refueling canal floor to permit flooding above the vessel during refueling. Since the leak-before-break concept is applied to the reactor coolant piping, the possibility of the local pressurization in the reactor cavity is eliminated. It becomes possible to permanently install the refueling pool cavity seal (termed as pool seal). Thus the need for assembling and disassembling temporary pool seals is eliminated. The current blind flanges are installed with bolts to seals off the containment building transfer tube penetration sleeve during reactor operation. Therefore, it takes long time to lift, and to temporarily store it. The QOBF, however, can reduce work load need for lifting it from the area during refueling.

It is expected that these design features shorten refueling work as many as about 3 days. So for the KNGR, it would take 42 days for refueling/maintenance (R/M) cycle per 18 months (i.e., 28 days/year). While current domestic nuclear power plants are operated with 12～18 month R/M cycle, KNGR is being designed to have a capability of operating over 18 month fuel cycle, from post refueling startup to the subsequent post refueling startup as per EPRI URD.

An extended outage is another type of outage. It is extended by the works such as ISI (In Service Inspection) and Turbine/Generator overhaul. Ultrasonic Test of the reactor vessel is performed according to the 10-year ISI program. Containment ILRT and turbine/generator overhaul are assumed to perform every 5 years and Steam Generators replacement is assumed once during a 60-year plant lifetime. Although the quantitative estimation of their effects are difficult, the extended outage is estimated as 6 days per year based on the operating experience of YGN 3 and Kori Unit 1. Thus the total annual planned outage was estimated about 34 days per year.

Ⅲ. Effects on the Plant Safety

The shutdown safety model of KNGR PSA consists of initiating events and recovery information based on generic industry data as well as plant specific data such as YGN 3 refueling perspectives. The existing at-power faults trees have been modified and enhanced to consider shutdown operation configurations. There are some reasons why plant risk during shutdown is comparable to that during plant operation. One is that some of systems and their combinations are placed out of service for extended maintenance: some of electrical power and decay heat removal are disabled, for example. Another is that reactor coolant inventory may be greatly reduced to the mid-loop for refueling. Finally, the reactor system, various other systems and the containment may be aligned open to the atmosphere. Refueling mode of the plant will bring about more radiation-exposing state, though level of decay heat is considerably low.

In the course of developing KNGR outage milestone schedule, some outage impacts by KNGR design features are identified. For KNGR, the Shutdown Cooling System (SCS) overhaul schedule in Figure 2 can be removed on the schedule. The overhaul of SCS of YGN 3&4 can be performed only in the condition of the refueling cavity fully filled during refueling. On the other hand, the KNGR SCS with backup of containment spray system (CSS) enables the system do without Technical Specification (T/S) Limiting Condition for Operation (LCO) violation during the power operation mode unlike the conventional SCS, which makes on-line maintenance possible. It is anticipated that such an effect brings much benefit to the KNGR outage management and shutdown safety. The RAM analysis in section Ⅱ indicated that overhaul duration could be significantly reduced by using the advanced design features such as IHA, PPS, and QOBF. It will contribute to the shutdown safety at the T/S operation mode 5 which is still need careful control for decay heat removal.

LOOP(Loss of Offsite Power) is an important initiating event during shutdown operation. Since one of the two Emergency Diesel Generators (EDGs) is out of service according to the T/S mode 4 through 6, the only one engineered safety feature train including EDG can be operable as a mitigation action after LOOP event. If both emergency diesel generators are available during shutdown operation, the plant risk due to LOOP which is almost take up more than 50% at this time will be greatly decreased. To avoid maintenance activity for emergency diesel generators during shutdown operation, Technical Specification relaxation for on-line maintenance of at-power condition is recommended.

Figure 3 shows a KNGR mid-loop operation scheme. Reduced inventory is defined as having the reactor coolant level of 3 feet below the flange or lower and mid-loop operation that the coolant is reduced to the mid-point of the hot leg so that nozzle dam can be installed. The KNGR has extensive instrumentation for reduced inventory operation and also has improved piping to lower the chance of cavitation of the SCS pumps. But the most significant improvement should be implemented in procedures and operator training to align the SCS and CSS system for configuration control. Also aggressive valve testing and maintenance program on the SCS and CSS would reduce the shutdown risk.

There are some initiating events which can lead to potential core damage. Almost all the initiating events cause plant trip in the accident sequences. The results of RAM study provides information to reduce initiating event frequencies and thus to enhance plant safety at power operation mode. The results shows that secondary steam removal systems including turbine/generator are major contributes for trip causes of korean nuclear power plant. Even if it also shows the reducing trend of trip frequency in recent years, still more concern is needed to further reducing the transients by secondary system failures.

Ⅳ. Conclusion and Discussion

The plant availability factor during the lifetime is expected to meet the goal of 90%. The trip frequency is expected to be less than 0.8 per year based on trip experiences of the domestic nuclear power plants and KNGR design concepts. In addition to availability, by reducing trip frequency and planned outage duration, the it is assured to enhance safety at both power operation and shutdown modes. Although main purpose of the RAM program is to ensure the plant availability, it will be further developed with closer relationship to the safety during the next detailed design phase. This integrated process will be checked by Reliability Assurance Program(RAP) which can be applied in the on-going development of KNGR to ensure the safety and availability evaluated in PSA and RAM.

References

1. "Annual Trip Reports for Nuclear Power Plants", KEPCO, 1979～1996
2. "The Third Evaluation Report for the KNGR availability", KEPCO, February 1999
3. KNGR SSAR, Volume 17, Chapter 19.8 "Shutdown Risk Assessment", KEPCO, 1999

Table 1. System Level Plant Trip Frequencies

System	Trips	Frequency (/year)	Percentile	Remarks
T/G and Auxiliaries	71	0.86	31.8 %	- Total Trips
Feedwater System	40	0.49	17.9 %	: 223
Electrical Power System	38	0.46	17.0 %	- Average Trip Frequency
CEDM	23	0.28	10.3 %	: 2.71/year
Reactor Coolant System	18	0.22	8.1 %	- Total Operating Years
Subtotal	190	2.30	85.2 %	: 82.34 years

Table 2. Component Level Plant Trip Frequencies

Component/Subsystem	Trips	Frequency (/year)	Percentile
Turbine	26	0.32	11.7 %
Generator	26	0.32	11.7 %
Feedwater Control Valve	23	0.28	10.3 %
CEDM Control Circuit	17	0.21	7.6 %
Off-site Power/Switchyard	15	0.18	6.7 %
Turbine Governor Valve	8	0.10	3.6 %
Transformer	8	0.10	3.6 %
Feedwater Pump	8	0.10	3.6 %
Electrical Bus	8	0.10	3.6 %
Main Steam Isolation Valve	7	0.09	3.1 %
Subtotal	146	1.77	65.4 %

Table 3. Root Cause and trend of the Plant Trip

Description	Trip Frequency(/year)		
	During All the Period('78 ~ '96)	During All the Period (excluding first one year after commercial operation)	During Recent 5 Years ('92 ~ '96)
Functional Failure	2.47(75.5%)	1.85(76.7%)	0.95(71.7%)
Human Error	0.56(17.2%)	0.36(14.8%)	0.29(20.0%)
External Event	0.16(5.0%)	0.14(5.8%)	0.05(3.3%)
Others	0.08(2.3%)	0.06(2.7%)	0.07(5.0%)
Total	3.27	2.71	1.47

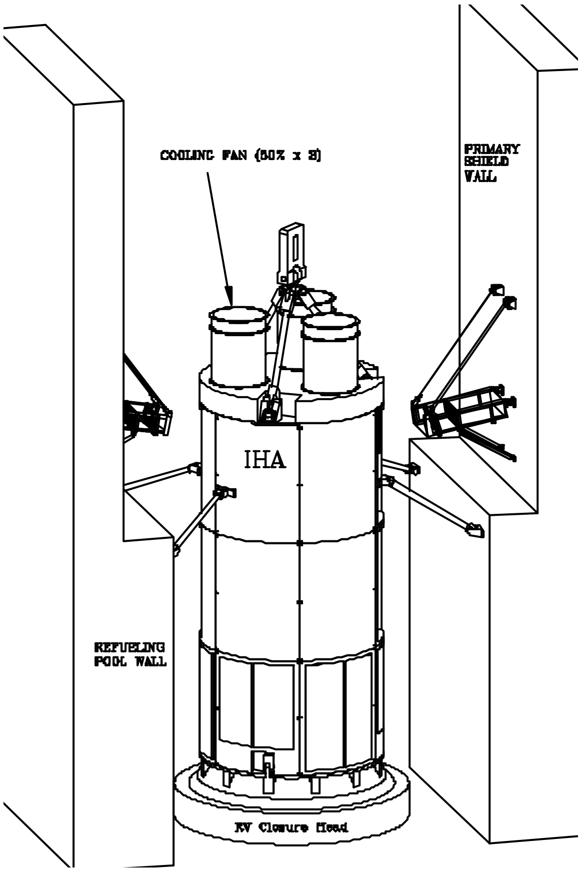

Figure 1. Integral Head Assembly

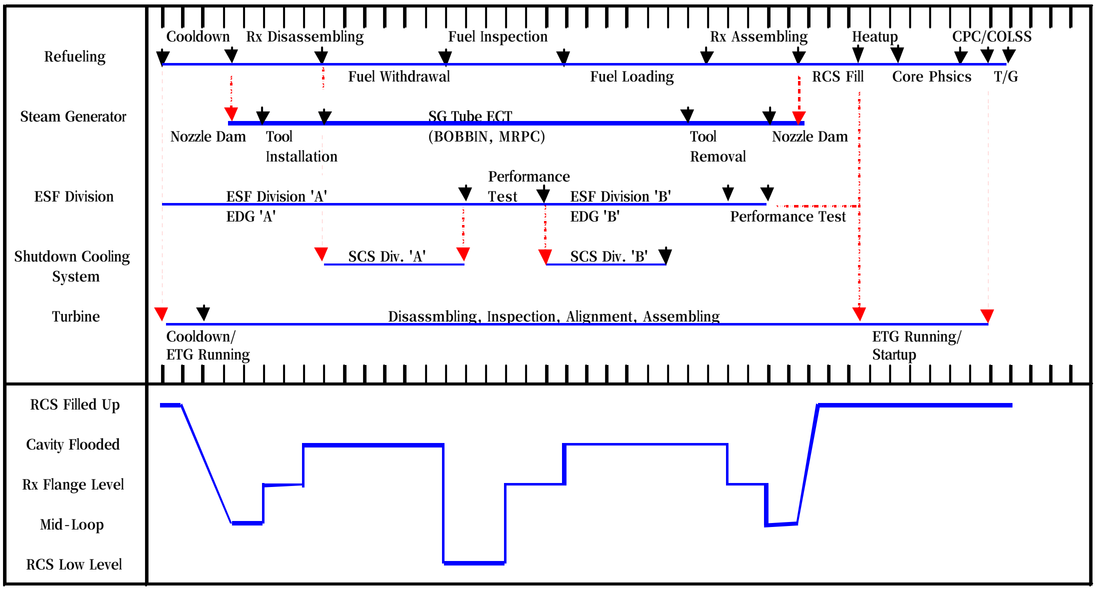

Figure 2. Refueling and Maintenance Outage Schedule

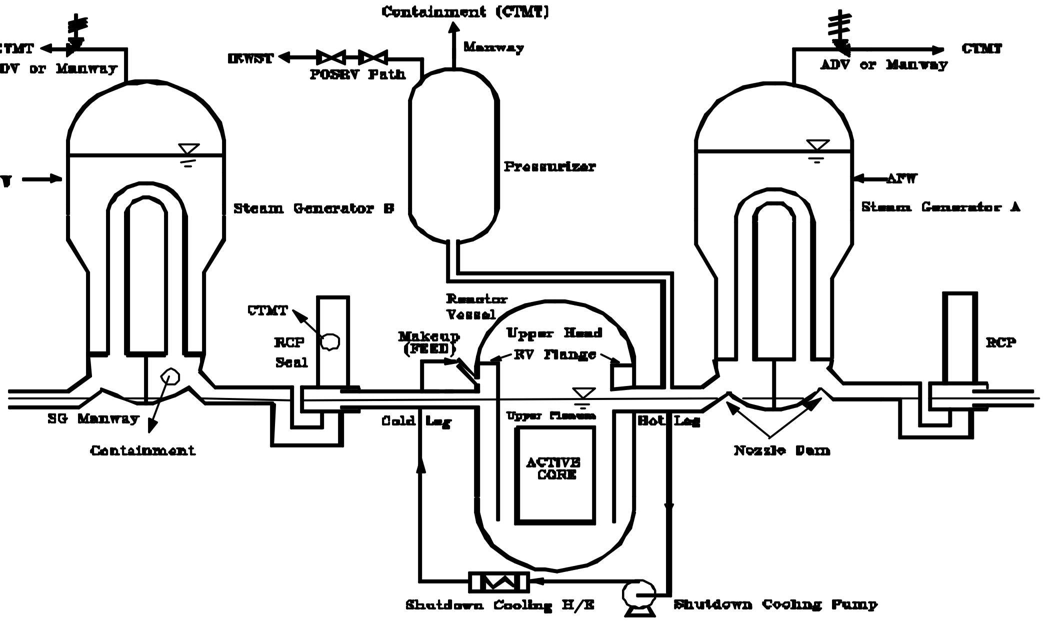

Figure 3. Mid-loop Operation Scheme