Transactions of the 15th International Conference on
Structural Mechanics in Reactor Technology (SMiRT-15)
Seoul, Korea, August 15-20, 1999

The Development of the Advanced Light Water Reactor in Korea -The Korean Next Generation Reactor-

Han Gon Kim, Young Sheop Park, Byong Sup Kim and Sung Jae Cho

Korea Electric Power Research Institute, Korea

ABSTRACT

Korean next generation reactor (KNGR), which is to be designed as a standard evolutionary advanced light water reactor (ALWR) in Korea, has been developed from 1992 as one of long-term government projects. The major characteristics of the KNGR are as follows; KNGR is 2-loop PWR and its design life time is 60 years. The CDF and the CFF will be much lower than 10^{-5}/RY and 10^{-6}/RY, respectively. For the design improvement, KNGR adopted inconel-690 as a steam generator tube material, four train ECCS, refueling water storage tank inside containment, and double cylindrical concrete containment. For more reliable and easier control, compact workstations have been adopted in the design of main·control complex and digital I&C technology is used for protection, control, and monitoring. In addition, KNGR has some passive design features such as fluidic device in safety injection tank, passive secondary condensing system, and passive auto-catalytic hydrogen recombiner to enhance safety.

1 Introduction

Since 1980's, many nuclear power plant vendors have developed advanced light water reactors, which are safer and more economic than current nuclear power plants. The typical examples of ALWR are System 80+ by ABB-CE, ABWR by General Electric, and EPR by NPI, and so on. In Korea, the KNGR program being continued since 1992 to prepare for the domestic nuclear power demand expected early in the 21st century, is on the line extended from the present Korea standard nuclear power plant (KSNP) in an overall view.

We developed the design goals and characteristics for KNGR in the first phase of the project. The major design goals are as follows;

- The reactor type is 2-loop evolutionary PWR.
- The rated thermal and electrical power are 4000MW and 1450MW, respectively.
- The design life time is 60 years.
- Core damage frequency (CDF) and containment failure frequency (CFF) are less than 10^{-5}/RY and 10^{-6}/RY, respectively.
- The thermal margin is greater than 10 %. The target is 15%.
- The economic goal is to secure around 20% cost advantage over competing energy sources, for example, coal fired power generation.
- The KNGR envelops not only all the sites in Korea but also almost sites in the world.

To meet above goals, KNGR adopted following design features;

- Double cylindrical concrete containment,

- Four train emergency core cooling system which is injected to the reactor vessel directly,
- In-containment refueling water storage tank (IRWST),
- Improved steam generator material and design,
- Main control room designed under the human factors engineering principles and digital I&C technique,
- Fluidic device in safety injection tank in order to regulate flow rate in passive manner,
- Passive secondary condensing system (PSCS) to remove decay heat in passive manner, and
- Severe accident mitigation system including passive auto catalytic recombiner and cavity flooding system.

In second phase of the development, the basic design has been performed to meet the above design goals and characteristics. Currently, we finished KNGR basic design at Feb. 1999.

In this paper, we describe the design status of the KNGR focused on nuclear systems.

2. Description of the nuclear systems

The nuclear steam supply system of the KNGR is designed to operate at rated output of 4000 MWth to produce an electric power output of around 1450 MWe. The major components of the primary circuit are a reactor vessel, two coolant loops, each containing one hot leg, two cold legs, one steam generator (SG), and two reactor coolant pumps (RCPs), and one pressurizer (PZR) connected to one of the hot legs.

The core consists of 241 fuel assemblies with an average enrichment of 2.6 w/o in a 16x16 array. The number of CEAs is 93 with 8 spare CEAs. 17 of the 93 CEAs are part-strength CEAs. The absorber materials used for full-strength control rods are boron carbide (B_4C) pellets. Inconell alloy 625 is used as the absorber material for the part-strength control rods. The core is designed for an operating cycle of 18-24 months, and has an increased thermal margin of up to 15 % to enhance safety and improve operation performance. A portion of the fuel rods contains uranium fuel mixed with a burnable absorber (Gadolinium) to suppress excess reactivity after fuelling and to help control the power distribution in the core.

The reactor is comprised of a vertically mounted cylindrical vessel with a hemispherical lower head welded to the vessel and a removable hemispherical closure head. The major design improvements incorporated in the reactor design include; larger operating margins, higher power level, and lower failure rate of fuel elements for higher plant availability and reliability. The life time of the reactor pressure vessel is improved to 60 years by use of low carbon steel, which has lower contents of Cu, Ni, P, S compared to current designs. The core support structures are designed to support and orient the reactor core fuel assemblies and control element assemblies, to direct the reactor coolant to the core. The core support barrel and the upper guide structure are supported at its upper flange from a ledge in the reactor vessel flange. The flange thickness is increased to sustain the enhaned seismic requirements.

In addition, KNGR adopted integrated head assembly (IHA) for availabilty improvement by reduction outage time. IHA is a mechanical assembly of various components required to provide lifting the reactor vessel closure head and its appurtenances, cooling of the CEDM, supporting the head area cables and protecting missiles generated from the reactor vessel head area. One piece removal of head assembly is estimated to save almost three days of refueling outage time.

The design temperature in the hot leg is reduced to 615 F which is lower than that of the currently operating plants in order to increase the operating margin and to reduce the SG tube corrosion problem. The capacities of the PZR and the SGs (especially secondary side) are increased from that of current designs. The increased capacity of the pressurizer accomodates the plant transients without power operated relief valves. Conventional spring loaded safety valves mounted to the top of the PZR are replaced by the pilot operated safety relief valves (POSRVs), and functions of the RCS overpressure protection and safety depressurization could be performed by the POSRVs. The increased water inventory on the secondary side reduces the pontential for unplanned reactor trips and provide longer operator response time in case of the total loss of feed water accident.

The power control system is capable of daily load follow operation with frequency control operation at a typical load variation profile in Korea; 16 hours at 100 % and 4 hours at 50% with 2 hours ramps for power decreases and increases. To do this, Mode K algorithms are under development which controls core power automatically according to power demand. The load rejection capability at the rated power should also be incorporated. This capability can reduce the outage time caused by the secondary system troubles since the reactor power can be brought up to 100% as soon as the troubles have been fixed.

3. Safety systems and features

3.1 Safety goals and design philosophy

One of the KNGR development policies is to increase the level of safety dramatically. To implement this policy, the plant will be designed in accordance with the established licensing design basis to meet the licensing criteria and also be designed with an additional safety margin in order to improve the protection of the investment, as well as the protection of the public health. The safety goals of the KNGR can be summarized as follows;

- The total core damage frequency should not exceed 10^{-5}/yr, considering both internal and external initiating events.
- The whole body dose for a person at the site boundary should not exceed 0.01 Sv (1 rem) during 24 hours after initiation of core damage even in the event of containment failure. The probability exceeding such a limit should be less than 10^{-6}/yr.
- The frequency of an accident in which the release of long-lived radioisotopes such as Cs-137 would exceed the amount to limit the land use shall be less than 10E-6 per year.

In addition to the public safety, a concept of investment protection will be implemented in the KNGR design. In KNGR, there are many investment protection goals such as loss-of-coolant-accident (LOCA) protection, steam generator inventory, and so on. For example, the reactor with its fuel should be used continuously following the event of small break LOCA upto 15 cm pipe break.

Another important design philosophy for safety is the increased design margins. A few examples of the design requirements following this philosophy are the requested core thermal margin of 10~15%, sufficient system capacity for operator recovery action time of more than 30 minutes, and station blackout coping time of 8 hours.

3.2 Active safety systems

The active safety systems consist of the safety injection system (SIS), safety depressurization system (SDS), in-containment refueling water storage system (IRWST), auxiliary feedwater system (AFS), and containment spray system (CSS).

The main design concept of the SIS is simplification and redundancy to achieve higher reliability and better performance. The safety injection lines are mechanically 4 trains and electrically 2 divisions without the tie branch between the injection lines for simplicity and independence as shown in Fig. 1. Each train has one safety injection pump and one safety injection tank. The low pressure safety injection pumps (LPSIPs) are eliminated due to the adoption of direct vessel injection because safety injection water is not spilled out through the break in case of large cold leg break LOCA. By the elimination of LPSIPs, safety injection and residual heat removal functions were separated. Through the IRWST the current operation modes of high pressure, low pressure, and re-circulation can be merged into only one operation mode (i.e., safety injection). According to PSA results, CDF is reduced by half by the adoption of 4 train independent safety injection system.

The refueling water storage tank is located at the inside of the containment and the arrangement is made in such a way that the injected emergency cooling water can return to the IRWST. The susceptibility of the current refueling water storage tank to external hazard is lowered by locating it at the inside of the containment. The functions of IRWST are as follows; the storage of refueling water, a single source of water for the safety injection, shutdown cooling, and containment spray pumps, a heat sink to condensing steam discharged from the pressurizer for rapid depressurization if necessary to prevent high pressure core melt or to enable feed and bleed operation, and coolant supply to the cavity flooding system in case of severe accidents to protect core melt.

The AFWS is designed to supply feedwater to the SGs for RCS heat removal in case of loss of main/startup feedwater systems. In addition, the AFWS refills the SGs following a LOCA to minimize leakage through pre-existing tube leaks. The AFWS is a 2 divisions and 4 trains system. The reliability of the AFWS has been increased by use of two 100% motor-driven pumps, two 100% turbine-driven pumps and two independent safety-related emergency feedwater storage tanks as a water source instead of condensate storage tank.

3.3 Passive safety systems

KNGR is an evolutionary plant which relies on the active systems for ensuring its safety. However, some passive systems, such as fluidic device in safety injection tank (SIT) and the passive secondary condensing system (PSCS), are incorporated in the KNGR design.

During end of blowdown and refill phase of cold leg LBLOCA, large amount of water should be injected into the downcomer to fill the core. After the water is filled upto the level of cold leg, however, all of additional water is spilled out to the break. It means that large amount of water can not be used to mitigate LOCA. Therefore we adopted fluidic device in SIT to regulate the amount of injected water passively. As shown in Fig. 2, when water level is higher than stand pipe suction, SIT discharges large amount water sufficient to fill the core. When the water level is lower than stand pipe suction, water flows through control port. It makes large voltex resistance and results small amount of water discharge. In this phase, SIT serves safety injection water as much as LPSIP capacity.

We have studied on passive secondary heat removal system, namely passive secondary condensing system, to cope with total loss of secondary heat sink transients. PSCS consists of a water storage tank located on the auxiliary building, a condenser located in the tank, and related pipings and valves. In KNGR, PSCS has two 40 MW condensers, one for each steam generator, to mitigate limiting event, total loss of feedwater accident. In the event of loss of auxiliary feedwater system, the PSCS takes inlet flow into the isolation condenser submerged in the condenser tank from the steam line and discharges outlet flow into the feedwater line. This connection can make it possible passive core cooling without any active component running by natural circulation.

3.4 Severe accidents prevention and mitigation

The measures to cope with severe accident are divided into two categories, prevention and mitigation.

Severe accident prevention features can be summarized as follows:

- increased design margin such as larger pressurizer, larger steam generators, and increased thermal margin,
- reliable ESF systems such as SIS, AFWS, and CSS,

- extended ESF systems such as SDS with IRWST, AAC, and PSCS, and
- containment bypass prevention.

Severe accident mitigation features and strategy can be summarized as follows;

- hydrogen mitigation system such as passive auto-catalytic recombiner and hydrogen ignitor,
- wide reactor cavity area and cavity cooling system,
- SDS and IRWST,
- robust double containment with large volume, and
- in-vessel retension of molten core.

4. Man machine interface system (MMIS)

KNGR is equipped with digitialized Man-Machine Interface System(MMIS) which encompasses the control room systems and Instrumentation and Control(I&C) systems, reflecting the modern computer technology.

One of the main features of the I&C system is the use of microprocessor-based multi-loop controllers for the safety including reactor protection and non-safety control systems. Engineering workstations and industrial personal computers are used for the two diverse data processing systems, respectively. To keep the plant safety against common mode failures in software due to the use of digital systems, controllers of diverse types and manufacturers will be employed in the control and protection systems. For data communication, a high speed fibre optic network is used. The remote signal multiplexer is also utilized for the safety and non-safety systems field signal transmission to save considerable amount of cables and cable trays. Since the S/W is heavily relied on in full digital MMIS, stringent S/W qualification process will be established and followed for the life cycle of KNGR. The MMIS concept to be implemented in the KNGR design is schematically depicted in Figure 3.

The KNGR MCR design is characterized by 1) redundant compact workstations for operators, 2) seismically qualified Large Display Panel(LDP) for overall process monitoring of the plant to be shared among operating crew 3) multi-functional soft controls for discrete and modulation control, 4) computerized procedure system to provide on one of the workstation CRTs with context sensitive operation guides, operational information, and navigation links to the soft controls for normal and emergency circumstances and 5) safety console for dedicated conventional miniature button type controls provided to control essential safety functions. CRTs and FPD(flat panel display) are extensively used for presentation of operational information.

The human factor engineering is an essential element of the control room facility design and MMI design and its principles are systematically employed to ensure safe and convenient operation. Operating experience review analysis, function analysis, and task analysis are performed to provide systematic input to the MMI design.

Partial dynamic mockup has been constructed based on the simulator of predecessor plant(KSNP) system models. This facility is used to perform initial verification of suitability of the MMI design. In the forthcoming design stage, the mockup will be expaned for intermediate validation of the design and I&C prototyping will be undertaken for smooth development of KNGR MMIS facility.

5. Balance of plant systems

5.1 General building arrangements

The general arrangement of KNGR has been developed based on the twin-unit concept and slide-along arrangement with common facilities such as the radwaste building as shown in Figure 4(a). The auxiliary building is designed to surround the containment building to achieve 4 train design concept for the safety systems. This type of auxiliary building prevents the direct access of the containment during construction. The auxiliary and containment buildings will be built on a common basemat. The common basemat will improve the resistance against seismic events and reduce the number of walls between buildings so that rebar and formwork cost can be reduced.

In addition to the current arrangement, KNGR has been considered to have complex building which consists of access control building, radwaste building, and hot machine shop as shown in Figure 4(b). The purpose of this building, which is located at the center of two plant, is to enhance operability by the effective facility management.

5.2 Containment

The containment building consists of a reinforced concrete outer containment, a steel-lined, post-tensioned concrete inner containment, and a reinforced concrete internal structure. The containment building is designed to provide biological shielding, external missile protection, and to sustain all internal and external loading conditions which may reasonably be expected to occur during the life of the plant. The equipment hatch, which has 7.8 m (26 feet) inner diameter, is selected to accommodate one-piece replacement of a steam generator. A polar bridge crane is supported from the containment wall. The bridge crane has the capability to install and remove the steam generators.

The outer containment is composed of a reinforced concrete right cylinder with a shallow, domed roof. It has an inner radius of 25.8 m (86 feet). Annular space, called the Annulus, is provided between the inner radius of the outer containment and the outer radius of the inner containment above the basemat. The main function of the Annulus is for collection of post-LOCA containment atmosphere leakage. This leakage is filtered, recirculated, and released by the annulus ventilation system. Adequate access is provided for installing, testing, inspecting, and tensioning the tendons.

The inner containment is a post-tensioned concrete cylinder with a hemispherical dome. There is no structural connection between the free standing portion of the inner containment and the adjacent structures other than penetrations and their supports. The lateral loads due to seismic and other forces are transferred to the foundation concrete through the structural concrete reinforcing connections. The containment free volume has been set at 9.1×10^4 m^3 (3.2×10^6 ft^3).

5.3 Turbine generator

The turbine generator systems are designed to be capable of operation at 3% house load for a period of at least 4 hours without any detrimental effects of the systems, and capable of startup to full load from cold conditions in 8 hours, including rotor preheat. The main steam lines and the high pressure turbine are designed for a steam pressure of 6.9 MPa (1,000 psia), and two reheater stages are provided between the high pressure and the low pressure turbines. The generator is a three phase, 4 pole unit operating at 1800 rpm.

The feedwater pump configuration is selected to be 3x50% because of its ability to allow more reliable operation; all three pumps are normally operating, and the plant can remain at 100% power operation if one of the feedwater pumps is lost. On-line condensate polishers which can operate in full and partial flow, as well as in bypass mode, are provided to maintain proper water chemistry during normal power operation. In the feedwater systems, the feedwater heaters are installed in 7 stages and arranged horizontally for easy maintenance and reliability.

5.4 Electrical systems

The main power system consists of the generator, the generator circuit breaker, the main transformer, the unit auxiliary transformer and the stand-by transformer. The normal power source for non-safety and permanent non-safety loads is the off-site power source and the generator. If the normal power source is not available, the permanent non-safety loads are covered by two alternative sources: one from the stand-by off-site power source (via the stand-by transformer) and the other from one non-1E alternate AC power source with a gas turbine generator.

The electric power necessary for the safety-related systems is supplied through 4 alternative ways:

1. the normal power source, i.e., the normal off-site power and the in-house generation,
2. the stand-by off-site power, i.e., the off-site power connected through the stand-by transformer,
3. the on-site standby power supply, i.e., two diesel generators, and
4. the alternative AC source, i.e., the gas turbine generator.

6. Conclusions

The KNGR development project was organized in three phases related to the development status. Phase I of the project was scheduled to run for the two-year period from the end of 1992 to the end of 1994, and the major activity was to develop top tier design requirements and concepts for the new design. Phase I was finished according to the plans, and is now followed by Phase II which is a four-year programme, running from 1995 to 1999. The major activities of this phase are to develop a basic design for a licensing review, to ensure the safety of the KNGR and thus, its licensibility. The level of design completion by the end of the Phase II is estimated to be around 20% of the total engineering works needed for construction, and commissioning, of a plant.

The basic design and safety analysis has been finished Feb. 1999. Now, we are undergoing Phase III programme. The major activities of this phase are design optimization to enhance economics and safety further and obtaining design certification. The phase III programme will be finished end of 2001.

7. References

[1] Final Reports for Research and Development on Next Generation Reactor (Phase I), Korea Electric Power Corporation, December 1994.

[2] Interim Reports for Research and Development on Next generation Reactor(Phase II), Korea Electric Power Corporation, March 1998.

[3] Sung Jae Cho and Dong Wook Jerng, "Research Activities and Design Requirements For the Next Generation Reactor in Korea", The International Workshop on Future LWRs, Tokyo, Japan, July 1995.

[4] Sung Jae Cho, Byung Sop Kim, and Myung Gie Kang, "Passive Design Features for Application to Korean Next Generation Reactor," POST-SMiRT 14 International Seminar "Passive Safety Features in Nuclear Installations" Italy, August, 1997

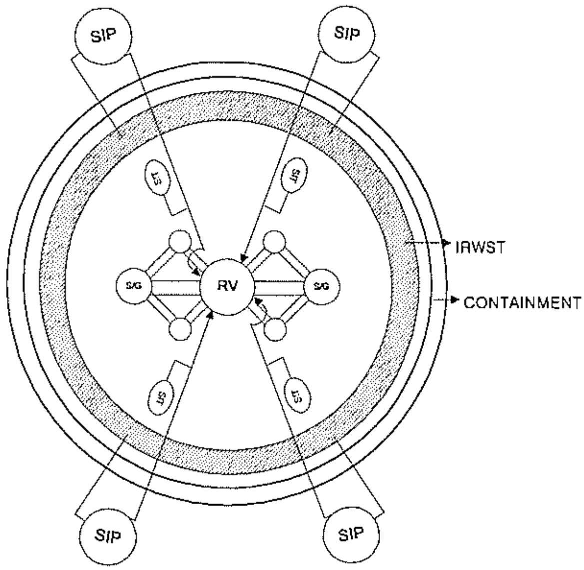


Fig. 1 Arrangement of KNGR safety injection system

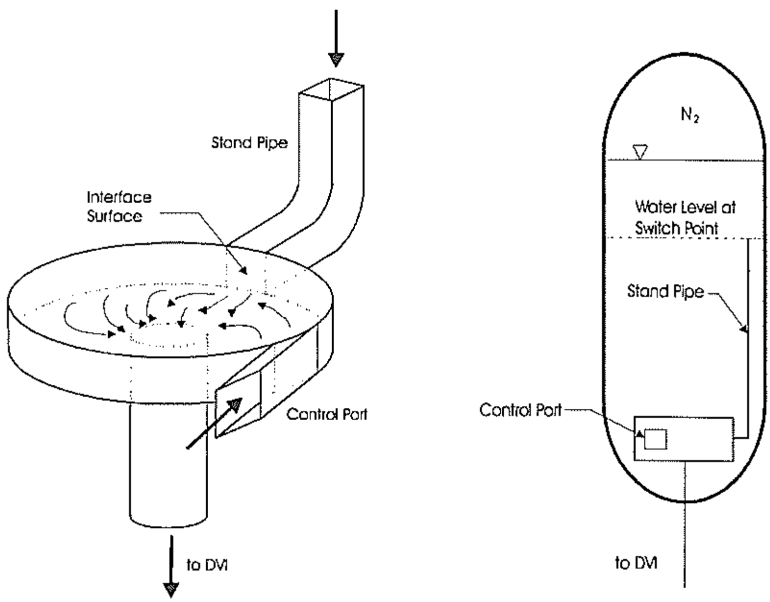


Fig. 2 Schematic Diagram of Fluidic Device in SIT

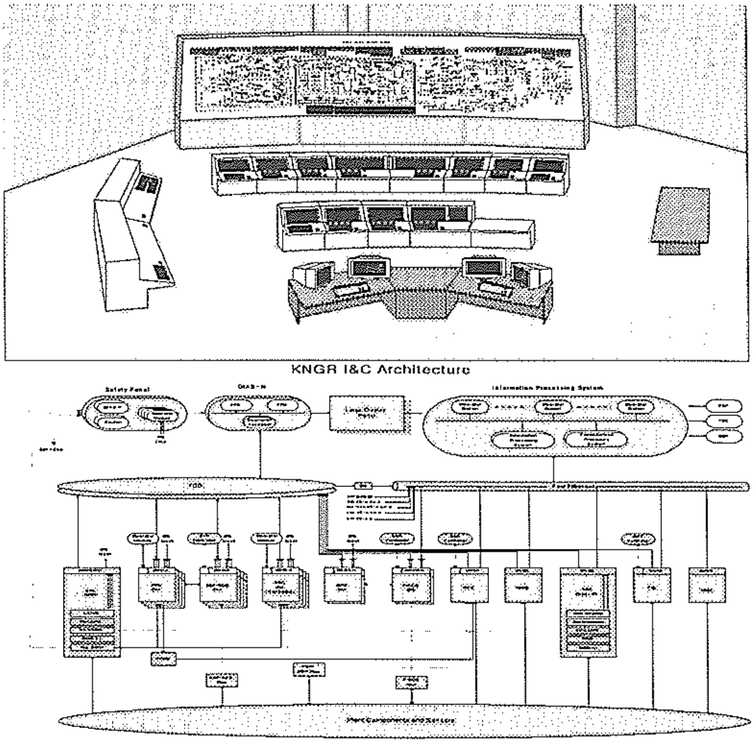
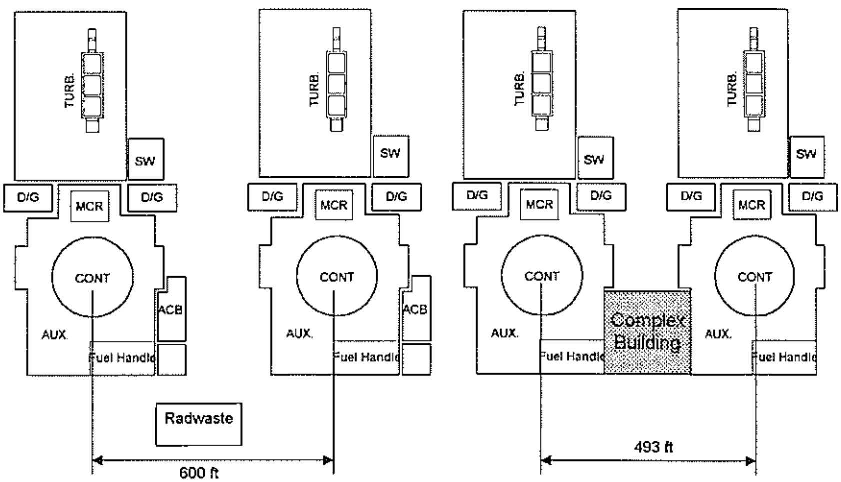


Fig. 3 KNGR man-machine interface system(MMIS)



(a) Standard general arrangement

(b) general arrangement with complex building

Fig. 4 The general building arrangement of KNGR