# 1. INTRODUCTION AND GENERAL DESCRIPTION OF PLANT

## 1.1 INTRODUCTION

This Final Safety Analysis Report (FSAR) is submitted in support of the application of the Korea Hydro & Nuclear Power Company(KHNP) to obtain an operating license for the Kori Nuclear units 3 and 4 (KRN 3 & 4).

### 1.1.1 THE KORI NUCLEAR UNITS 3 AND 4 390

Unit 3 and Unit 4 each consist of a pressurized water reactor, a turbine generator, and associated auxiliaries. The two units are virtually identical with a limited number of shared facilities. Kori Nuclear Units 3 and 4 will hereafter be referred to as the facility.

The facility is located at the Kori Nuclear Power Plant site which is on the southeastern coast of the Korean peninsula approximately 16 miles south of Ulsan and 20 miles northeast of Pusan.

The containment for the nuclear steam supply system (NSSS) is a dry, prestressed concrete, steel-lined structure which is designed by Overseas Bechtel, Incorporated (Bechtel).

The NSSS is manufactured by the Westinghouse Electric Corporation (Westinghouse). The NSSS is a pressurized water reactor (PWR) type.

The reactor core is designed for a thermal output of 2900 MWt. When the reactor coolant pumps heat input of 12 MWt is added to the core output, the resulting warranted NSSS output is 2912 MWt, which is defined as the rated power in the license application.

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The turbine generator is rated for operation at the NSSS warranted output of 2912 MWt with a corresponding electrical output of 1033.2 MWe. This corresponds to a net electrical output for each unit of 953.2 MWe. The turbine is supplied by General Electric Company of England (GEC) and the generator is supplied by Hitachi .Ltd. Power & Industrial systems.

The date of completion of construction and initiation of fuel loading at Unit 3 is anticipated to be April 1984 and the date of commercial operation is anticipated to be October 1984. The construction completion date and commercial operation date for Unit 4 are anticipated to be April 1985 and October 1985, respectively. 390

## 1.1.2 THE FINAL SAFETY ANALYSIS REPORT

### 1.1.2.1 Organization and Format

This FSAR for the KNU 5 & 6 adheres to Regulatory Guide 1.70, Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants, Revision 2, prepared by the Regulatory Staff of the U.S. Nuclear Regulatory Commission (USNRC) and issued in September 1975.

the FSAR is paginated to provide flexibility when incorporating changes to text and figures. All text pages are numbered by section; e.g., 1.1-1 is the first page of section 1.1. Tables and figures are also numbered in a similar manner; e.g., table 1.1-1 is the first table in section 1.1., and they are placed at the end of the section. In some cases, appendices are included at the end of a chapter of the FSAR to provide supplemental information. 357

Topical reports and other documents referenced in the text are listed at the end of each section. Topical reports and other documents incorporated into the application by reference are also listed in section 1.6, Material Incorporated by Reference.

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Standards used for abbreviations and symbols are the latest editions of the following American National Standards Institute publications: ANSI-Y1.1, Abbreviations; ANSI-Y10.19, Letter Symbols for Units Used in Science and Technology; and ANSI-Y10.5, Letter Symbols for Quantities Used in Electrical Science and Electrical Engineering. Abbreviations used in this FSAR are defined as they first occur.

When additional or revised information is incorporated in this FSAR, those pages affected will be identified with the amendment number in the lower corner of the margin opposite the binding margin, and the amendment date in the lower corner of the binding margin. A vertical line with the amendment number is placed in the binding margin next to the material affected.

### 1.1.2.2 Approach to Design Bases

As used within this FSAR, the design bases are a list of requirements that the system must meet in order to:

A. Perform directly a specific safety or power generation function, including support of another system in performance of its required function; e.g., provide cooling water flow for other components, maintain a given compartment temperature, etc.

B. Comply with a regulatory or statutory requirement or guideline; e.g., a regulatory guide or a jurisdictional building code.

C. Meet a specific operator interface, startup, or important or specific testing requirement.

D. Meet a design classification or code requirement; e.g., be designed to withstand the safe shutdown earthquake (SSE).

In effect, the design bases for a system are determined by assessment of the characteristics the system must have due to requirements imposed by other systems, regulations, guidelines, or other factors separate and distinct from the components of the system itself. The system itself is thought of as a unit, the boundaries of which are defined by Regulatory Guide 1.70, and which must meet specific requirements. The design bases should describe all essential characteristics of the system with sufficient clarity so that an experienced engineer, using these design bases and material referenced in the design bases, can understand the functions of the system with respect to the rest of the plant. Items implicit to contemporary design; e.g., use of the English system of weights and measures or the exercise of good engineering practice, are not specified.

1.1.2.2.1 Safety Design Bases

Safety design bases are those engineering objectives which directly establish or increase nuclear safety. Specifically, safety design bases provide for, or assure, the following:

A. The integrity of the reactor coolant pressure boundary (RCPB)

B. The capability to shut down the reactor and maintain it in a safe shutdown condition

C. The capability to prevent or mitigate the consequences of accidents that could result in potential offsite radiation exposures comparable to the guideline exposures of Title 10, U.S. Code of Federal Regulations, Part 100 (10 CFR 100)

D. The accomplishment of a specifically articulated requirement for structures, systems, or components important to safety.

The control room operator is considered as one of the fundamental means of achieving these criteria.

Safety-related equipment or structures, systems, and components important to safety means the portions of systems which are indispensable to nuclear safety. Items that are associated with safety-related equipment, but which in themselves are not absolutely essential to a nuclear safety function, are not safety-related.

Redundancy requirements and system performance conditions are considered a feature of the equipment's capability to shut down the reactor safely, or to prevent or mitigate accidents.

#### 1.1.2.2.2 Power Generation Design Bases

Power generation design bases are those design bases which are not related to nuclear plant safety. They are called power generation design bases because they necessarily relate directly or indirectly to power generation, in the sense that all station requirements which are not imposed for safety reasons support the major function of the station as a whole; i.e., the generation of electrical power. An example of a power generation design basis is the requirement to provide domestic water for plant personnel.

### 1.1.2.3 Drawings

Drawings are included throughout this FSAR with the appropriate system descriptions. Symbols and abbreviations used on Bechtel- and Westinghouse-designed flow diagrams and piping and instrumentation diagrams (P&IDs) are shown in figures 1.1-1 and 1.1-2, respectively. Tables 1.7-1 and 1.7-2 provide cross references between FSAR figure numbers and drawing numbers to facilitate the identification of pertinent material.

### 1.1.2.4 Definition

ITS is an acronym of 'Improved Technical Specification', "운영기술지침서" in Korean, which is used in FSAR as follows:

a. ITS Chapter 1 3.4 is 운영기술지침서 제1편 3.4 357

b. ITS Chapter 1 Table 3.3.1-1 is 운영기술지침서 제1편 Table 3.3.1-1

ITS Base is "운영기술지침서 기술배경서", used as follows:

a. ITS Base B 3.4 is 운영기술지침서 기술배경서 B 3.4

KOREA ELECTRIC POWER CORPORATION
KOREA NUCLEAR UNITS 5 & 6
FSAR

SYMBOLS & LEGEND
BOP FLOW DIAGRAMS AND
P&I DIAGRAMS
(Sheet 1 of 3)

Figure 1.1-1

본 도면은 참고용 도면임
관련 도면목록 :Table 1.7-1(Sh. 1 of 12)
관리도면 : P&ID , 0-M-XX-F011

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Amendment 539
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KOREA HYDRO & NUCLEAR POWER
COMPANY KORI 3 & 4 FSAR

SYMBOLS & LEGEND
BOP FLOW DIAGRAMS AND
P&I DIAGRAMS
(SHEET 2 OF 3)
Figure 1.1-1

KOREA ELECTRIC POWER CORPORATION
KOREA NUCLEAR UNITS 5 & 6
FSAR

SYMBOLS & LEGEND
BOP FLOW DIAGRAMS AND
P&I DIAGRAMS
(Sheet 3 of 3)

Figure 1.1-1

KOREA ELECTRIC POWER CORPORATION
KOREA NUCLEAR UNITS 5 & 6
FSAR

NSSS FLOW DIAGRAM
LEGEND

Figure 1.1-2

## 1.2 GENERAL PLANT DESCRIPTION

### 1.2.1 SITE DESCRIPTION

#### 1.2.1.1 Site Location

The Kori site is on the southeastern coast of the Korean peninsula 16 miles south of Ulsan and 20 miles northeast of Pusan at the village of Kori on a point known as Kodang-Mal. The approximate coordinates are [redacted] Figure 1.2-3 shows the general location of the station.

The Kodang-Mal point is approximately one-half-mile wide and extends southward about one-half mile from the main coastline. The Kori Nuclear Power Station is located at the southern tip of the point. The topography of the point is typical of the area topography which is characterized by mountains and valleys and a fairly sheer coastline. Elevations in the site region range from about 20 feet above msl to 262 feet above msl at the exclusion boundary. In grading the site to provide level areas for the plant structures, the cut was used as fill to form a strip at the end of the point 3575 feet long by 900 feet wide. Units 1 and 2 are located on the western section of the strip at elevation 20 feet while Units 5 and 6 are on the eastern section of the strip at elevation 31 feet. **The Kori 345kV switchyard is located north-west of the strip at elevation 248 feet.** 532

#### 1.2.1.2 Geology

Geologically, the site lies within the Cretaceous-aged Milyang basin and has an extremely complex geologic structure. On an areal basis, almost all of the site is underlain by ancient sedimentary rock formations which have been folded and faulted. The rock layers and faults are intersected and intruded by many igneous intrusive dykes. No area within the plant site is free of faults or dykes.

The faults and dykes are ancient features which have not moved in at least 10 to 15 million years and do not form a major fault system. Although the area is inactive seismically, earthquakes, ranging in intensity to a maximum of JMA (Japan Meteorological Agency Scale) IV to V, have occasionally been reported within the site region. At the site area, the maximum historical intensity of an earthquake has been estimated at JMA III.

The Kodang-Mal Point is surrounded by a narrow, irregular, gravelly and rocky beach which gives way to a sandy beach along the northwest shore at the villages of Kilchon-Ri and

Wolnae-Ri. At a distance of 60 to 90 feet inland from the shoreline, the topography rises abruptly to about 26 feet above sea level and then flattens somewhat, but continues to rise northward across the site. At about 65 feet elevation, the surface grade increases and continues to rise northward until it reaches the crest of a steep hill at 285 feet above sea level. The small but prominent bluff which parallels the shoreline is cut at several locations by minor surface drainage.

#### 1.2.1.3 Meteorology

The site is on the southeast coast of Korea and the climate can be described as continental in nature during the winter season. The strong summer weather systems are influenced primarily by tropical air masses from the south-southwest, while in the winter season, the weather is influenced by polar continental air masses.

Snow is uncommon during winter, with an estimated annual accumulation of 30 cm/y. Precipitation during the winter months occurs principally as rain on the southeast coast. Heavy precipitation occurs during the summer months and is most often associated with the passage of typhoons and tropical storms which occur primarily in the months of May through September.

An average of two typhoons a year come close enough to the coast to affect Korea. These typhoons usually bring torrential rainfall which, combined with high tides, results in flood conditions for low-lying areas along the coast. However, less than one typhoon actually crosses the Korean coast.

Average annual rainfall is 1300 mm with 40 to 50 percent of the rainfall occurring during the months of June through August. Maximum hourly rainfall is 68 to 71 mm. Flooding will not occur at the site because of its location and topographical features.

The high that develops off the Pacific Ocean during the summer season results in a moist southerly flow of air from the South Sea to the site region.

During the winter season, a semi-permanent high pressure cell develops over the central region of Mongolia, resulting in a prevailing northwesterly flow of air into the site area. The mean annual wind speed for the Pusan area is 4.4 m/s. The mean annual wind speed for the Ulsan area and the site area is 2.7 m/s and 3.6 m/s respectively.

The maximum recorded temperature was 38C at Ulsan. The record minimum was -16.7C. The normal annual relative humidity from Pusan and Ulsan is 66 percent and 71 percent respectively.

The recorded highest temperature was 33.5C at the site.

### 1.2.1.4 Hydrology

The KNU 5 & 6 site is on the southeastern coast of Korea, approximately 32 kilometers northeast of Pusan.

High and low water levels, as indicated by the land datum, are as follows:

| | |
|---|---|
| Highest high-water level | + 1.133 m |
| High water level | + 0.705 m |
| Mean water level | + 0.309 m |
| Low water level | - 0.087 m |
| Lowest low-water level | - 0.497 m |

### 1.2.1.5 Population Distribution

The 1981 residential population within 16 kilometers of the site was 128,225. This is shown in table 1.2-1 and figure 1.2-1. About 42,190 people live within 8 kilometers of the site.

The area within 16 kilometers of the plant site is predominantly rural and is characterized by mountains and hills, interspersed with farmlands. The 1981 residential population between 16 kilometers and 80 kilometers of the site was 5,660,600. This is shown in table 1.2-2 and figure 1.2-2.

Population densities vary by sectors; relatively higher density sectors are found in the southwestern sectors bordering the seashore.

### 1.2.2 PLANT ARRANGEMENT

The Korea Nuclear Units 5 and 6 (KNU 5 & 6) are comprised of two pressurized water reactor nuclear steam supply systems (NSSS) that will have a production per unit of 993 MWe at maximum core power. The station features separate containment, auxiliary, control and component cooling water buildings, turbine buildings, diesel generator buildings, fuel storage buildings, auxiliary buildings, and shared intake structure, radwaste building, auxiliary boiler, and material storage facilities. The ultimate heat sink for all Seismic Category I cooling water systems is salt water from the East Sea, supplied to the component cooling water heat exchangers by nuclear service cooling water pumps located within the intake structure. Seawater pumped from the intake structure by the circulating water pumps serves as the heat sink for the heat rejected by the main condensers and the turbine plant cooling water system. The turbine generator orientation with respect to the containment is peninsular and the two units are in a slide-along orientation. **The Kori 345 kV switchyard is located north-west** of the two units. The plant arrangement is presented in figures 1.2-3 through 1.2-16. 532

### 1.2.3 NUCLEAR STEAM SUPPLY SYSTEM

For each unit, the NSSS consists of a pressurized water reactor (PWR), reactor coolant system (RCS), and associated auxiliary systems. The RCS (figure 1.2-17) is arranged as three reactor coolant loops connected in parallel to the reactor vessel, each containing a reactor coolant pump and a steam generator. An electrically heated pressurizer is connected to the hot leg of one reactor coolant loop.

High pressure water circulates through the reactor core to remove the heat generated by the nuclear chain reaction. The heated water exits from the reactor vessel and passes via the coolant loop piping to the steam generator. Here it gives up its heat to the feedwater to generate steam for the turbine generator. The cycle is completed when the water is pumped back to the reactor vessel. The entire RCS is composed of leaktight components to ensure that all fluids are confined to the system.

The core is of the multi-region type. All fuel assemblies are mechanically identical, although the fuel enrichment is not the same in all assemblies. In the initial core loading, three fuel enrichments are used. Thirty-six fuel assemblies with the highest enrichment and sixteen assemblies with the second highest enrichment are placed in the core periphery, or outer region. The other sixteen assemblies with the highest enrichment are located inboard on the core diagonals. Thirty-six assemblies with the second highest enrichment and fifty three assemblies with the lowest enrichment are arranged in a selected pattern in the central region (see figure 4.3-1). In subsequent refuelings, approximately one-third of the fuel is discharged and fresh fuel is loaded into the core. The remaining fuel is arranged in the core in such a manner as to achieve optimum power distribution.

Rod cluster control assemblies (RCCAs) are used for reactor control and consist of clusters of cylindrical absorber rods. The absorber rods move within guide tubes in certain fuel assemblies. Above the core, each cluster of absorber rods is attached to a spider connector and drive shaft, which is raised and lowered by a drive mechanism mounted on the reactor vessel head. Following trip, downward movement of the rod cluster control is by gravity.

The reactor coolant pumps are Westinghouse vertical, single-stage, centrifugal pumps of the shaft-seal type.

The steam generators are Westinghouse vertical U-tube units which contain Inconel tubes. Integral moisture separation equipment reduces the moisture content of steam to 0.25 percent or less.

The reactor coolant piping and all of the pressure-containing and transfer surfaces in contact with reactor water are stainless steel clad, except the steam generator tubes which are Inconel and the fuel tubes which are Zircaloy. Reactor core internals, including control rod drive shafts, are primarily stainless steel.

An electrically heated pressurizer connected to one reactor coolant loop maintains RCS pressure during normal operation; limits pressure variations during plant load transients and keeps system pressure within design limits during abnormal conditions.

Auxiliary system components are provided to charge the RCS and add makeup water, purify reactor coolant water, provide chemicals for corrosion inhibition and reactivity control, cool system components, remove decay heat when the reactor is shut down, and provide for emergency safety injection.

### 1.2.3.1 Principal Design Criteria

The inherent design of the pressurized water, closed-cycle reactor significantly reduces the quantities of fission products which are released to the atmosphere. Four barriers exist between the fission product accumulation and the environment. These are the uranium dioxide fuel matrix, the fuel cladding, the reactor vessel and coolant loops, and the containment system. The consequences of a breach of the fuel cladding are greatly reduced by the ability of the uranium dioxide lattice to retain fission products. Escape of fission products through fuel cladding defects would be contained within the reactor pressure vessel, coolant loops, and auxiliary systems. Breach of these systems or equipment would release the fission products to the containment where they would be retained. The reactor containment is designed to adequately retain these fission products under the most severe accident conditions, as analyzed in chapter 15.

Design criteria are established to ensure the following:

A. The minimum departure from nucleate boiling ratio during normal operation and anticipated operational occurrences which provides at least a 95% probability with 95% confidence that departure from nucleate boiling does not occur, is not less than the limit value described in Subsection 4.4.1.1. 382

B. The maximum fuel centerline temperature evaluated at the design overpower condition is below that value which could lead to centerline fuel melting. The melting point of the $UO_2$ is not reached during normal operation and anticipated operational occurrences.

C. Fuel rod clad is designed to maintain cladding integrity throughout fuel life. Fission gas release within the rods and other factors affecting fuel design life are considered for the maximum expected exposures.

D. The reactor system is designed so that any xenon transients are adequately damped.

E. The RCS is designed and constructed to maintain its integrity throughout the expected plant life. Appropriate means of test and inspection are provided.

F. Power excursions that could result from any credible reactivity addition incident do not cause damage either by deformation or rupture of the pressure vessel, nor do they impair operation of the engineered safety features (ESF).

G. The combined response of the fuel temperature coefficient, the moderator temperature coefficient, the moderator void coefficient, and the moderator pressure coefficient to an increase in reactor thermal power is a decrease in reactivity. In addition, reactor power transients remain bounded and damped in response to any expected change in any operating variable.

#### 1.2.3.2 Operating Characteristics

The reactor is controlled by temperature coefficients of reactivity; by control rod cluster motion, which is required for load follow transients and for startup and shutdown; and by the soluble neutron absorber, boron, in the form of boric acid. The boric acid is inserted during cold shutdown, is partially removed at startup, and is adjusted in concentration during core lifetime to compensate for such effects as fuel depletion and accumulation of fission products which tend to slow the nuclear chain reaction.

The control system allows the plant to accept step load increases of 10 percent and ramp load increases of 5 percent per minute over the load range of 15 to 100 percent of full power. Equal step and ramp load reductions are possible, over the range of 100 to 15 percent of full power.

Below 10 percent power, the unit can accept a turbine trip without initiating a reactor trip.

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#### 1.2.3.3 Safety Considerations

Several ESFs have been incorporated into the plant design to reduce the consequences of a loss of coolant accident (LOCA). These safety features include a safety injection system which automatically delivers borated water to the reactor vessel for cooling the core under high and low reactor coolant pressure conditions. The safety injection system also serves to insert negative reactivity into the core in the form of borated water during an uncontrolled plant cooldown following a steam line break or an accidental steam release. Other safety features which have been included in the reactor containment design are a containment air recirculation and cooling system which acts to limit the containment pressure following a loss of coolant, and a containment spray system which acts to depressurize the containment and remove elemental iodine from the atmosphere.

The reactor is refueled by equipment which handles spent fuel under water from the time it leaves the reactor vessel until it is placed in a shipping cask for shipment from the site. Underwater transfer of spent fuel provides an economic and transparent radiation shield, as well as a reliable coolant for removal of decay heat.

The fuel handling system is divided into two areas: the refueling canal area, which is flooded for refueling; and the fuel pool, which is external to the containment and is always accessible to plant personnel. The two areas are connected by a fuel transfer system which carries the fuel through an opening in the containment.

Spent fuel is removed from the reactor vessel by a refueling machine and is placed in the fuel transfer system. In the fuel pool, the fuel is removed from the transfer system and is placed into storage racks. After a suitable decay period, the fuel is able to be removed from storage and loaded into a shipping cask for removal to a reprocessing plant.

Primary coolant leakage from the reactor vessel closure seals, pressure and relief valves, pump and relief valve seals, and other identified leakage sources of primary coolant are collected at the source and are piped to the equipment drain tank or pressurizer relief tank. This decreases the radioactivity released to the containment atmosphere and increases the sensitivity of detecting unidentified leakage.

The nuclear design analyses and evaluation establish physical locations for control rods and burnable poisons and physical parameters such as fuel enrichments and boron concentration in the coolant. The nuclear design evaluation established that the reactor core has inherent characteristics which, together with corrective actions of the reactor control and protective systems, provide adequate reactivity control even if the highest reactivity worth RCCA is stuck in the full withdrawn position.

The thermal-hydraulic design analyses and evaluation establish coolant flow parameters which assure that adequate heat transfer is provided between the fuel clad and the reactor coolant. The thermal design takes into account local variations in dimensions, power generation, flow distribution, and mixing. The mixing vanes incorporated in the fuel assembly spacer grid design induce additional flow mixing between the various flow channels within a fuel assembly as well as between adjacent assemblies.

Instrumentation is provided in and out of the core to monitor the nuclear, thermal-hydraulic, and mechanical performance of the reactor and to provide inputs to automatic control functions.

## 1.2.4 ENGINEERED SAFETY FEATURES AND EMERGENCY SYSTEMS

Engineered safety features (ESFs) protect the public and plant personnel in the highly unlikely event of an accidental release of radioactive fission products from the reactor system, particularly as the result of a LOCA. These safeguards function to localize, control, mitigate, and terminate such accidents to hold exposure levels below 10 CFR 100 limits.

### 1.2.4.1 Containment System

#### 1.2.4.1.1 Containment Building

The containment is a prestressed, reinforced concrete structure in the shape of a cylinder with a hemispherical roof and a flat foundation slab. The cylindrical portion of the containment structure is prestressed by a post-tensioning system consisting of horizontal (hoop) and vertical (inverted U) tendons. Hoop tendons are anchored at three buttresses equally spaced around the containment structure. These tendons extend 240 degrees around the cylinder periphery, bypassing intermediate buttresses. The dome has a two-way, post-tensioning system. Prestressing of the hemispherical dome is achieved by a two-way pattern of tendons that are a part of the vertical (inverted U tendons). The dome horizontal tendons start at the springline and continue up to the 45 degree line of the dome.

The foundation slab is conventionally reinforced with high-strength reinforcing steel. Near the outer edge of the foundation slab, a continuous access gallery is provided for installation and inspection of vertical tendons. The interior surface of the containment shell is steel-lined for leaktightness. A protective layer of concrete covers the portion of the liner over the foundation slab. The containment structure concrete provides biological shielding for normal and accident conditions.

The containment building completely encloses the reactor and RCS and is designed so that the leakage of radioactive materials to the environment will not exceed 0.1 percent volume in 24 hours, even in the unlikely event of a LOCA. The dimensions of the containment are as follows: inside diameter, 130 feet; inside height, 195 feet above filler slab; cylindrical wall nominal thickness, 4 feet, and minimum dome thickness at the top of the dome, 3 feet 3 inches. The net free volume is approximately 2,080,000 $ft^3$. The internal structures and compartment arrangement provide equipment missile protection and biological shielding for maintenance personnel. 5

The containment building is designed for all credible loading combinations, including normal loads, loads during a LOCA, test loads, and loads due to adverse environmental conditions. The following are considered:

A. Loads caused by the pressure and temperature transients due to accident conditions

B. Dead loads

C. Thermal Loads

D. Live loads

E. Earthquake loads

F. Wind loads

G. Typhoon loads

H. External pressure loads

I. Prestressing loads

J. Missile impact loads

The containment building design pressure of 60 psig is greater than the peak pressure that would occur as a result of the pipe break accidents analyzed in subsection 6.2.1. Energy contribution from the steam system, due to reverse heat transfer through the steam generator tubes, is included in the calculation of the design containment pressure transient. The supports for the RCS are designed to withstand the forces associated with the sudden severance of the reactor coolant piping with coincidental earthquake. In addition, the containment design pressure will not be exceeded during any subsequent long term pressure transient as determined by the combined effect of heat sources such as residual heat and limited metal-water reactions, structural heat sinks, and the operation of the ESF using only emergency onsite electrical power.

The containment building is discussed further in section 6.2.

#### 1.2.4.1.2 Containment Spray System

The Containment Spray System (CSS) is provided to cool the atmosphere and reduce the pressure in the containment. It also provides chemical additive (sodium hydroxide) to remove iodine following a postulated LOCA. A common spray additive tank and two 100 percent capacity spray subsystems are provided, each consisting of a pump, spray header, and spray additive eductor. The containment spray pumps take suction initially from refueling water storage tank. The spray additive eductor takes suction from the spray additive tank.

The containment spray system is further discussed in subsections 6.2.2 and 6.5.2

1.2.4.1.3 Containment Atmosphere Emergency Cooling System

The containment atmosphere emergency cooling system (CAECS) is provided to cool the atmosphere and reduce the pressure in the containment. Four 100 percent atmosphere coolers are provided.

The containment atmosphere emergency cooling system is further discussed in section 6.2.

1.2.4.2 Emergency Core Cooling System (ECCS)

The principal ECCS mechanical components which provide core cooling immediately following a LOCA are the accumulators (one for each reactor coolant loop), the centrifugal charging (high head safety injection) pumps, the residual heat removal (low head safety injection) pumps, and the associated valves, tanks, and piping.

In order that fuel rod damage, to the extent that it would impair effective cooling of the core, is prevented, the ECCS is designed to cool the reactor core and to provide additional shutdown capability following initiation of the following accident conditions:

A. Pipe breaks and valve failures in the RCS which cause a discharge larger than that which can be made up by the normal makeup system, up to and including the instantaneous circumferential rupture of the largest pipe in the RCS.

B. Rupture of a control rod drive mechanism (CRDM) causing a control rod assembly ejection accident.

C. Pipe breaks and valve failures in the steam system, up to and including the instantaneous circumferential rupture of the largest pipe in the steam system.

D. A steam generator tube rupture.

The acceptance criteria for the consequences of each of these accidents is described in chapter 15 in the respective accident analyses sections.

In order to ensure that the ECCS will perform its desired function during the accidents listed above, it is designed to tolerate a single active failure during the short term immediately following an accident, or to tolerate a single active or passive failure during the long term following an accident.

In the highly unlikely event of a LOCA, the centrifugal charging pumps and the residual heat removal pumps take suction from the refueling water storage tank and inject borated water into the RCS. This provides cooling to limit core damage and fission product release and ensures adequate shutdown margin. After the inventory of coolant in the refueling water storage tank is completely depleted, the borated water in the containment sump is recirculated through the residual heat removal heat exchangers and the reactor core to provide long term cooling of the core.

### 1.2.4.3 Emergency Electrical Systems

#### 1.2.4.3.1 Standby Power Supply

A diesel generator, complete with accessories, fuel transfer systems, and fuel storage system is provided as the standby power supply for all of the equipment of each of the two redundant ESF trains for each unit. Each diesel generator is rated at 7000 kW for continuous operation and 7700 kW for short time operation at 4.16 kV, 60 Hz. There are a total of four diesel generators for KNU 5 & 6.

Two independent and redundant air starters, actuated by air at 200 to 250 psig, are used to provide the torque to automatically start each diesel generator upon receipt of either an ESF actuation signal or an unacceptable degradation of the voltage on the 4.16 kV Class 1E bus to which the generator is connected. Each diesel generator is able to accept loads in approximately 10 seconds after the receipt of a starting signal and complete the starting sequence for all required safety-related equipment within 60 seconds thereafter.

A 7-day onsite inventory of fuel oil is provided for each diesel generator. Oil is transferred from the underground cylindrical storage tank to the day tank by redundant transfer pumps with the excess flow being returned to the storage tank.

The heat of combustion is transferred from the engine cylinder liners, the fuel injectors, the lubrication system, and the intercooler by a closed cycle cooling water system heat exchanger to the nuclear services cooling water system. The cooling water is circulated by a pump driven by the diesel engine.

Ventilation and combustion air for each diesel generator room is provided by two 50 percent capacity, direct drive, vaneaxial exhaust fans at a total flowrate of 150,000 cfm and a static pressure of 1.0 in. wg. When the diesel is not operating, the ventilation flowrate is 6300 cfm, which is provided by a direct drive, vaneaxial exhaust fan.

1.2.4.3.2 Emergency Lighting System

The emergency lighting system provides an adequate amount of dc lighting in the areas used during shutdown or emergency. These areas include the main control room, the auxiliary shutdown panels, and local control stations required to shut down and maintain the plant in a hot shutdown condition from outside the control room.

In other critical areas and emergency exit corridors where the minimum illumination required for the safety of plant personnel is furnished. The dc lighting consists of strategically located and seismically qualified portable battery pack units. The fixtures are designed to provide rated lighting for a minimum continuous period of 8 hours.

1.2.4.4 Fire Protection System

Fire protection is achieved at the station through the application of fire prevention, fire detection, and fire extinguishment methodologies.

Noncombustible and heat resistant materials are used in the station construction to minimize the potential for the occurrence of fires and to limit the flame intensity if a fire should occur. Sufficient separation between the components of redundant safety systems is provided to maintain the integrity of at least one of the systems so that the plant may be shut down safely. Where sufficient separation cannot be achieved, fire barriers are used between the two safety systems to ensure system integrity.

A combination of fire and smoke detection equipment is used throughout the station to detect the occurrence and location of a fire. Ionization chambers are used to detect combustion products. Photoelectric cells and ionization chambers are used to detect smoke. Infrared sensors are used to detect the presence of heat and flames. Heat actuated devices are used to detect the energy released by the fire. The selection of a sensor for a specific application is determined by an evaluation of the fire potential for each area of the station.

The fire extinguishment systems include at least the following major features:

A. Fire protection water supplies, yard mains, hydrants, and valves

B. Automatic wet sprinklers, hydraulically designed, ordinary hazard schedule

C. Water spray systems, hydraulically designed

D. Automatic pre-action sprinklers, ordinary hazard

E. Standpipes and hose reels

F. Portable extinguishers

The water inventory in the two 435,000 gallon storage tanks is delivered to the sprinkler, spray, and standpipe systems by two 1500 gpm automatic diesel engine-driven and one 1500 gpm motor-driven fire pump at a pump discharge pressure of 125 psig. Through the use of main loops, each branch is provided with two sources of water.

Selected components of the fire protection system are designed to meet Seismic Category I criteria in order to provide manual fire fighting capability in safety-related areas in the event of safe shutdown earthquake. This system is designed to supply water from Seismic Category I Condensate Storage Tanks to at least two standpipes and hose connections in areas containing equipment required for safe shutdown.

The fire and smoke monitoring, detection, and alarm system includes a supervisory circuit which indicates the failure of individual circuits and detectors. Both monitoring and supervisory alarm signals register locally and on the audible-visual fire annunciator panel in the control room.

### 1.2.4.5 Control Room Emergency HVAC System

The control room emergency HVAC system is designed to maintain the control room habitable during the postulated conditions resulting from a hypothetical LOCA, a fuel handling accident, or any other occurrence generating high level airborne radioactivity at the control room air intake.

The emergency HVAC system consists of two redundant trains whose major components include a supply/recirculation air handling unit and a filtration train. Upon receipt of an ESF actuation signal, fuel building high airborne radiation signal, or control room outside air intake high radiation signal, the control room emergency HVAC system is automatically started. The emergency HVAC system may also be initiated manually from the control room.

Transfer to the emergency mode consists of isolating the habitability area from the normal HVAC system, opening the outside air isolation valve to the emergency filtration trains, and starting the fans in the emergency filtration train and in the emergency supply/recirculation unit. Should, for some reason, the self-closing doors to the control room be prevented from shutting, an alarm reminds the operator to shut them. The emergency train fan discharges into the emergency supply/recirculation air handling unit. Thus, the emergency train fan draws outside air through HEPA filters and carbon adsorbers and discharges into the control room; since there is no control room exhaust, the control room atmosphere exfiltrates to the outside of the control room.

Control room air is recirculated at a rate of 21,000 cfm through the emergency supply/recirculation air handling unit to be cooled. The processing of outside air through the emergency filtration train allows 1000 cfm of outside air to be introduced into the control room. The temperature of the control room is maintained between 68 and 78F.

In addition, 1000 cfm of the recirculated air is processed through the emergency filtration train to aid in the removal of particulates and iodines which have built up in the control room atmosphere.

#### 1.2.4.6 Containment Isolation System

The containment isolation system consists of the piping, valves, and actuators required to isolate the containment following a LOCA, steam line rupture, main feedwater line rupture, or fuel handling accident inside the containment.

Each line which penetrates the containment and is either a part of the reactor coolant pressure boundary, or is connected directly to the containment atmosphere and is not a closed system, is provided with one of the following containment isolation valve arrangements.

A. One locked shut isolation valve inside and one locked shut isolation valve outside containment

B. One automatic isolation valve inside and one locked shut isolation valve outside containment

C. One locked shut isolation valve inside and one automatic isolation valve outside containment (a simple check valve is not used as the automatic isolation valve outside containment)

D. One automatic isolation valve inside and one automatic isolation valve outside containment (a simple check valve is not used as the automatic isolation valve outside containment)

Isolation valves outside containment are located as close to the containment as practical and, upon loss of actuating power, air-operated automatic isolation valves fail shut.

Containment isolation during a LOCA is initiated in two phases, phase A and phase B. Phase A containment isolation occurs upon any one of the following conditions:

A. Low pressurizer pressure

B. High containment pressure (Hi-1)

C. Low pressure steam line

D. Manual safety injection signal (SIS) actuation

E. Manual containment isolation actuation.

A phase A containment isolation signal isolates all process lines penetrating containment which do not increase the potential for damage of containment equipment when isolated and which are not required for operation of the ESF.

Phase B isolation is initiated upon sensing of containment high pressure (Hi-3). A phase B isolation signal isolates process lines penetrating containment which were not isolated during phase A and which are not required for operation of the ESF. These lines are those which supply cooling water and seal water flow to the reactor coolant pumps. Coincident with phase B isolation, the containment spray system is actuated.

Upon failure of a main steam line or main feedwater line, the containment isolation system isolates the containment and steam generators to prevent excessive cooldown of the RCS or overpressurization of the containment.

Steam line isolation is initiated upon receipt of one of the following signals:

A. High steam pressure rate

B. Low pressure steam line

C. Containment high pressure (Hi-2)

D. Manual actuation.

Upon initiation of steam line isolation, the main steam isolation valves and feedwater isolation valves are shut in order to prevent excessive cooldown of the RCS.

Feedwater line isolation is initiated upon receipt of one of the following signals:

A. Low pressurizer pressure

B. High containment pressure (Hi-1)

C. Low pressure steam line

D. Steam generator level Hi-Hi

E. Manual SIS actuation.

Upon initiation of the feedwater isolation signal, the feedwater isolation valves are shut in order to prevent excessive cooldown of the RCS.

The main steam isolation valves and piping are designed to prevent uncontrolled blowdown from more than one steam generator. The main steam isolation valves will shut fully within 5 seconds after steam line isolation is initiated. The blowdown rate is restricted by steam flow restrictors in each steam generator. For main steam line breaks upstream of an isolation valve, uncontrolled blowdown from more than one steam generator is prevented by the isolation valves in the unaffected steam lines and by the isolation valve in the affected line. For main steam line breaks downstream of an isolation valve, double blowdown is prevented by the main steam isolation valves on each main steam line.

Failure of any of the above components relied upon to prevent uncontrolled blowdown of more than one steam generator will not permit a second steam generator blowdown to occur. Piping restraints and pipe whip barriers between the main steam lines prevent a rupture in one line from causing a blowdown from more than one steam generator. No single active component failure will result in the failure of more than one main steam isolation valve to operate.

### 1.2.4.7 Combustible Gas Control System

A hydrogen recombiner system and a hydrogen purge system are used to control the concentration of hydrogen in the containment atmosphere to less than 3.5 percent following a postulated LOCA. Two independent, full capacity intake and return piping systems are provided at each unit, to which one or both of the mobile hydrogen recombiners that are shared by the two units may be connected within 24 hours after the start of the LOCA. The containment atmosphere is circulated through each recombiner at a rate of 70 cfm, which is sufficient to decrease the expected hydrogen concentration of 3.5 percent at 12 days after accident initiation to 3.3 percent at 20 days after accident initiation for the operation of one recombiner or to 2.6 percent for the operation of two recombiners.

If both of the mobile hydrogen recombiners should fail to perform their function, the hydrogen purge system may be valved into the recombiner intake piping system to provide a flow from the containment atmosphere of 50 cfm, which is compensated by a 50 cfm inflow from the compressed air system. The purge flow is passed through a mist eliminator, electrical heater, high efficiency air filter, a charcoal absorber, and a second high efficiency air filter prior to being released to the environment.

## 1.2.5 INSTRUMENTATION, CONTROL, AND ELECTRICAL SYSTEMS

### 1.2.5.1 Instrumentation and Control Systems

The reactor is controlled by temperature coefficients of reactivity; by control rod cluster motion, which is required for load follow transients and for startup and shutdown; and by a soluble neutron absorber. Boron in the form of boric acid which is inserted during cold shutdown, is partially removed at startup, and is adjusted in concentration during core lifetime to compensate for such effects as fuel consumption and accumulation of fission products which tend to slow the nuclear chain reaction.

The control system allows the plant to accept step load increases of 10 percent and ramp load increases of 5 percent per minute over the load range of 15 to 100 percent of full power. Equal step and ramp load reductions at 5 percent per minute are possible, over the range of 100 to 15 percent of full power.

Additional information on instrumentation and controls is presented in chapter 7.

#### 1.2.5.1.1 Introduction

To preclude unsafe conditions for equipment or personnel, the plant protection system monitors selected plant parameters in order to initiate reactor trip and/or ESF actuation. Multiple independent channels monitor each of the selected plant parameters. The plant protection system logic is designed to initiate automatically protective action whenever the monitored parameters reach a limiting safety system setting. Redundancy is provided in all parts of the plant protection system to assure that no single failure will prevent protective action when it is required.

The plant protection system is designed in conformance with IEEE Standard 279, Criteria for Protection Systems for Nuclear Power Generating Stations.

Sufficient redundancy is installed to permit periodic testing of the plant protection system so that a single failure or removal from service of any one protection system component or portion of the system does not preclude protective actions when required.

#### 1.2.5.1.2 Reactor Trip System Summary Description

The reactor trip system (RTS) includes the reactor protection system and the arrangement of components that perform the protective action after receiving a signal from the protection system or the licensed operator. The reactor trip signal deenergizes the control rod drive motor (CRDM) coils, allowing the RCCAs to drop into the core.

The instrumentation and controls of the RTS are specifically designed to permit on-line testing of the circuitry, down to the final actuating element, while the remaining portion of the system continues to conform to the single failure criteria. The instrumentation and controls are designed such that no single failure will initiate spurious reactor trip.

#### 1.2.5.1.3 Engineered Safety Features System Instrumentation Summary Description

The ESF systems include the engineered safety features actuation system (ESFAS) and the arrangement of components that perform protective actions after receiving a signal from the ESFAS or the operator.

The instrumentation and controls of the ESF systems are designed to permit testing while retaining the required capability for accomplishing their protective functions. Independence of redundant instrumentation and controls is provided so that a single failure within the systems will not prevent the ESF systems from accomplishing their protective functions.

#### 1.2.5.1.4 Systems Required for Safe Shutdown Instrumentation Summary Description

The systems required for safe shutdown (SRSS) are those systems which are required to shut down the reactor and maintain the reactor in a safe shutdown condition.

The instrumentation and controls of the SRSS are designed to permit testing while retaining the required capability for accomplishing their safe shutdown function.

Separation and independence of redundant instrumentation and controls is provided so that a single failure within the system will not prevent the SRSS from accomplishing their required functions.

#### 1.2.5.1.5 Safety-Related Display Instrumentation Summary Description

The safety-related display instrumentation (SRDI) provides information to enable the operator to perform the required safety functions.

The instrumentation provided monitors conditions in the RCS, the containment, the process systems, the RTS, the ESF systems, and the systems required for safe shutdown throughout planned operations, anticipated operational occurrences, and accident and post-accident conditions.

### 1.2.5.2 Electrical Power Systems

#### 1.2.5.2.1 Transmission and Generation Systems

The KEPCO grid system consists of interconnected hydroelectric plants, fossil fuel plants, and nuclear plants supplying electric energy over a system of 345 and 154 kV transmission lines and lower voltage distribution networks. The grid system is described in chapter 8.

KEPCO provides all of the electric power in South Korea. No interconnections are made with any other utility.

The main generator is an 1800 rpm, three-phase, 60-hertz, synchronous unit. The generator is connected through a coupling to the turbine shaft. 360

Power from the generator is stepped up from 22 to 362 kV by the unit main transformers and is supplied by overhead lines to The Kori 345 kV switchyard to both units. This switchyard is connected to the offsite transmission network by eight 345 kV transmission lines. 532

1.2.5.2.2 Electric Power Distribution Systems

Electric power is supplied from the switchyard to the onsite power system for the electrical auxiliaries of each unit through two independent circuits. Power is supplied to auxiliaries at 13.8 kV, 4.16 kV, and 480V levels.

The power distribution system includes Class 1E and non-Class 1E ac and dc power systems. The Class 1E power system supplies equipment used to shut down the reactor and limit the release of radioactive material following a design basis accident (DBA).

The Class 1E ac system for each unit consists of two independent and redundant load groups and four independent 120V vital ac instrumentation and control power supply systems. The load groups include 4.16 kV switchgear, 480V load centers and motor control centers (MCCs). The vital ac instrumentation and control power supply systems include battery systems, static inverters, regulating transformers, and distribution panels. Load group power voltage levels are designed for a range of plus/minus 10 percent. Vital ac instrumentation and control power systems are designed for a voltage range of plus/minus 2 percent. All equipment is furnished to be compatible with these voltage levels.

One independent diesel generator is provided as a standby source for each of the two Class 1E ac load groups of each unit. Each generator has sufficient capacity to operate all the equipment of one unit which is necessary to prevent undue risk to public health and safety in the event of a DBA.

One Non-Class 1E diesel generator which is in the independent AAC Building separated from the existing power blocks is provided as an alternate AC(AAC) source for common use of KORI Units 1,2,3 and 4 to cope with station blackout(SBO). This diesel generator is capable of supplying one division of shutdown loads of any Unit of KORI Units 1,2,3 and 4 which are required to bring the plant to hot shutdown mode and to maintain hot stand-by mode. 314

The non-Class 1E ac system includes 13.8 kV switchgear, 4.16 kV switchgear, 480V load centers, and MCCs.

Direct current power for the Class 1E dc loads of each unit is supplied by four independent Class 1E 125V batteries and associated battery chargers. One 250V and one 125V non-Class 1E battery and associated battery chargers supply power for the non-Class 1E dc system loads. And another direct current power for coping with station blackout(SBO) of KORI 1.2.3 and 4 is supplied by an independent Non-Class 1E 125V battery with an associated battery charger and a distribution panel. 314

These systems are discussed individually in chapter 8.

### 1.2.6 POWER CONVERSION SYSTEM

The steam and power conversion system removes heat energy from the reactor coolant in three steam generators and converts it to electrical energy with the turbine generator. These systems are discussed in detail in chapter 10.

The turbine generator is an 1800 rpm, tandem-compound, six-flow exhaust, condensing unit design for saturated steam conditions. The turbine section consists of one high-pressure turbine element and three low-pressure turbine elements. The generator rating is 1222 MVA at 22,000 volts and 60 hertz.

Steam enters the high-pressure turbine through the main stop valves and the governing control valves. From the high-pressure turbine exhaust, the steam flows through combined moisture separator/reheaters to the low-pressure turbines.

The main steam lines are provided with turbine bypass valves which discharge to the condenser, and safety, dump, and relief valves which discharge to the atmosphere.

Steam is taken from one of the moisture separator/reheaters for steam supply to the steam generator feedwater pump turbine drivers. The auxiliary feedwater pump turbine is supplied with driving steam from the main steam lines upstream of the main steam isolation valves. The two-stage reheaters are supplied with heating steam extracted from the high-pressure turbine and also from the main steam cross-connection header.

A turbine bypass system is provided which will bypass up to 36 percent of full-load main steam flow directly to the main condenser, and exhaust 28 percent of full-load main steam flow to the atmosphere. The bypass and atmospheric dump systems are designed to help maintain secondary side steam pressure below safety valve setpoint pressures during transient conditions. 321

The main condenser transfers unusable heat in the turbine exhaust steam to the water being circulated through the condenser tubes by the circulating water pumps. The circulating water is then returned to the East Sea.

Parallel strings of regenerative feedwater heaters utilize turbine extraction steam and turbine exhaust steam to heat the condensate and feedwater after it is pumped from the main condenser hotwell and before it is pumped into the steam generator.

### 1.2.7 FUEL HANDLING AND STORAGE SYSTEMS

The reactor is refueled by equipment which handles spent fuel under water from the time it leaves the reactor vessel until it is placed in a cask for shipment from the site.
Underwater transfer of spent fuel provides an economic and transparent radiation shield, as well as a reliable coolant for removal of decay heat.

The fuel handling system is divided into two areas : the refueling canal, which is flooded for refueling; and the fuel storage pool and fuel transfer canal, which is external to the reactor containment and is always accessible to plant personnel. The refueling canal and fuel transfer canal are connected by a fuel transfer system which carries the fuel through a penetration from the reactor containment.

Spent fuel is removed from the reactor vessel by a refueling machine and is placed in the fuel transfer system. In the fuel storage pool, the fuel is removed from the transfer system and is placed in storage racks. After a suitable decay period, the fuel can be removed from storage and loaded into a shipping cask for removal to a reprocessing plant.

New fuel and spent fuel assemblies are stored in the fuel building. New fuel assemblies are stored in vertical racks in either the new fuel storage pit or in the spent fuel storage pool. The new fuel storage pit is a reinforced concrete pit which provides dry storage for 60 new fuel assemblies. New fuel assemblies can also be in the spent fuel storage pool prior to initial fuel load or refueling. 308

169, The stainless steel lined reinforced concrete spent fuel storage pool
56, provides wet storage for 2,260 fuel assemblies (14.4 reactor cores) for Kori
15 3, and 2,262 fuel assemblies (14.4 reactor cores) for Kori 4. 308
Spent fuel assemblies are stored in vertical racks so spaced as to preclude criticality in a non-borate cooling water environment.
The design criteria and evaluation of the fuel storage facility are discussed in subsection 9.1.2. Control of the fuel storage pool water temperature during normal operation is accomplished by circulating the fuel pool water through a heat exchanger cooled by the CCWS. Purification and clarification of the fuel storage pool water is by the use of a filter, strainers, and an ion exchanger. A description and evaluation of the

fuel pool cooling and purification system is presented in subsection 9.1.3.

The fuel handling system, as further discussed in subsection 9.1.4, provides for the safe handling of fuel and control rod assemblies and for the required assembly, disassembly, and storage of reactor internals. These systems include a refueling machine located inside the containment above the refueling canal, the fuel handling crane, the fuel transfer carriage, the upending machines, the fuel transfer tube, a fuel handling machine in the spent fuel storage area, and various devices used for handling and storing the reactor vessel head and internals.

## 1.2.8 COOLING WATER AND OTHER AUXILIARY SYSTEMS

### 1.2.8.1 Cooling Water Systems

Water systems in operation at the facility include the nuclear service water system, CCWS, demineralized water system, domestic water system, condensate storage facility, reactor makeup water storage, refueling water storage, circulating water system, essential chilled water system, and central chilled water system, turbine plant closed cooling water system, and turbine plant open cooling water system. Except for the circulating water system, which is discussed in subsection 10.4.5, all of the water systems are covered individually in section 9.2.

The CCWS, essential chilled water system, refueling water storage system, condensate storage system and nuclear service water system are required for safe shutdown of the plant following a DBA, and the single failure criterion of 10 CFR 50 applies. These systems are Seismic Category I and each is designed for redundancy, as required, with functional and physical separation of each train of redundant components.

The CCWS is provided to remove heat from various reactor auxiliary systems which require cooling water of higher quality than service water. The system also provides a monitored intermediate barrier between potentially radioactive water and the service water system, reducing the probability of leakage of radioactive water into the service water.

The CCWS routes cooling water to equipment essential for normal operations of the NSSS and for cooling following a DBA. It also furnishes cooling water to various space cooling units throughout the containment.

Component cooling water circulates through two interconnected loops which are isolated to provide two physically separated closed loops following an ESF actuation. Each of these loops removes heat from the residual heat removal (RHR) heat exchangers, the containment spray pump lube oil cooler, the RHR pump lube oil coolers, the charging pump lube oil coolers, and the containment fan coolers. The CCWS also cools nonsafety-related auxiliary heat loads which function during normal plant operation. These nonsafety-related loads are automatically isolated from the safety-related loops following an ESF actuation. In these loops the temperature of water supplied to equipment being cooled varies according to loop bulk thermal load.

The nuclear service cooling water system furnishes cooling water from the ultimate heat sink for the component cooling water heat exchangers, the diesel generator cooling water heat exchangers, and the essential chilled water system. The system is composed of four nuclear service cooling water pumps and piping forming two physically separated nuclear service cooling water loops. Either of the nuclear service cooling water loops will satisfy service water requirements for safe shutdown of the plant under postulated DBAs.

The refueling water storage system is designed to supply borated water to the containment spray system and to the emergency core cooling system during the injection phase following a LOCA. The system consists of a 540,000 gallon tank and associated piping.

The essential chilled water system is designed to maintain the ambient temperature of the ESF equipment rooms and switchgear room during operation under accident conditions below maximum design ambient temperature. The major components of the system are the chiller, chilled water pump air separator, and compression tank.

The ultimate heat sink is required for the dissipation of residual heat after reactor shutdown and after an accident. The ultimate heat sink is the East Sea, which guarantees an unlimited cooling water supply in the event of an accident.

The other water systems of the plant are necessary for proper functioning of the plant. These auxiliary systems provide water as designated by their names and are analyzed in section 9.2.

### 1.2.8.2 Other Auxiliary Systems

The process auxiliary systems include the compressed air system, the process sampling system, the equipment and floor drainage system, the chemical and volume control system (CVCS), and the gross failed fuel detection system. Process auxiliaries are discussed further in section 9.3.

The compressed air system provides a reliable continuous supply of filtered, dried, oil-free air for pneumatic instruments and controls. The system also provides service air to outlets throughout the plant for pneumatic tools and other service requirements. The compressed air system is required for the plant normal operation; however, pneumatically operated devices in the plant which are required for safe shutdown are designed to move to the safe position upon loss of air pressure, or are provided with individual air accumulators or safety-related instrument air for emergency service.

The safety-related instrument air system consists of Seismic Category 1 air storage tanks, piping and valves. These tanks will automatically supply instrument air to the main steam power-operated relief valves and the flow control valves on the discharge of the motor-driven auxiliary feedwater pumps in the event that the normal instrument air supply is unavailable.

Process sampling systems are provided for the RCS and other secondary systems. Process sampling is used for determining chemical and radiochemical conditions of various fluids used in the plant.

The equipment and floor drainage systems are designed to collect drainage. The collected drainage from potentially radioactive equipment and floor drains can be processed in the liquid waste management systems. Monitoring capability is provided to ensure that inadvertent releases of radioactivity are prevented.

The CVCS is designed to provide a programmed water level in the pressurizer and the injection flow to the seal water system of the reactor coolant pumps. In addition, the CVCS controls water chemistry, reactivity level, concentration of soluble chemical neutron absorber, and makeup of the reactor coolant. The CVCS is also used for emergency core cooling.

The boron recycle system receives and recycles reactor coolant effluent for reuse of the boric acid and makeup water. The system decontaminates the effluent by means of demineralization and gas stripping, and uses evaporation to separate and recover the boric acid and makeup water.

The gross failed fuel detector is connected to the hot leg of a primary coolant loop. The coolant sample passes through a cooler and then into a coil containing a neutron detector and moderator, after which it flows back into the volume control tank. Chemistry samples can be taken to validate indications higher than normal. Rapid and large increases in activity are indicative of gross fuel failures.

### 1.2.8.3 Ventilation Systems Summary Descriptions

Ventilation systems are provided for normal plant operation and for DBA. The control building, access control building, turbine building, auxiliary building, fuel building, diesel generator rooms, radwaste building, nuclear service water pumphouse, and the containment are all provided with HVAC systems. These ventilation systems are designed to provide suitable environments for equipment and personnel. Where appropriate, ventilation zones are arranged within buildings to promote the flow of air from clean areas to areas of potentially greater radiation contamination prior to final exhaust. These systems are described further in sections 6.4 and 9.4.

The control building ventilation system maintains internal environmental conditions in the control room, cable spreading rooms, electrical rooms, and the computer room.

The control room ventilation system is designed to provide year-round air conditioning by filtering, heating, and cooling outside supply air and control room recirculated air. For post-DBA conditions, redundant emergency filtration trains are provided to maintain a habitable control room atmosphere. A smoke removal system is provided for purging.

The cable spreading room ventilation systems are designed to maintain temperatures compatible with normal operating requirements by filtering and cooling recirculated air from the cable spreading rooms. A smoke removal system is provided for purging.

The electrical room ventilation systems are designed to maintain temperatures compatible with the normal equipment operating requirements by filtering and cooling outside and recirculating air. Redundant systems for exhausting the battery rooms are provided for the removal of hydrogen.

The computer room and access control building ventilation systems are designed to maintain temperatures compatible with the working or operating equipment environment by filtering, heating, and cooling outside and recirculated air.

The fuel building ventilation system is designed to maintain a habitable working environment inside the building. Outside air is filtered, cooled, and supplied to the building through redundant supply units and is exhausted to outside atmosphere through redundant normal exhaust fans. An emergency fuel building exhaust system filters the building exhaust to mitigate the effects of postulated accidents.

The diesel generator rooms are ventilated to prevent equipment damage from extremes of temperature. A separate ventilation system is provided for each diesel generator room.

The auxiliary building ventilation system is designed to supply filtered and cooled outside and recirculated air to maintain temperatures compatible with normal operating requirements. Individual local recirculating units with cooling coils provide cooling for the ESF pumps during operation of the pumps, in addition to providing additional local cooling in selected areas. Air is exhausted through redundant filtered exhaust systems.

The turbine building ventilation system provides a source of outside air for removal of internally generated heat and heat transmitted to the building from external sources. Heated exhaust air is discharged to the atmosphere.

The containment ventilation system is designed for the following operating requirements:

A. Maintain temperatures at levels compatible with normal operating requirements by the operation of the containment fan coolers, reactor cavity cooling units, and CRDM shroud fans.

B. Remove heat from the containment atmosphere following a postulated LOCA or main steam line break inside the containment by the operation of the containment fan coolers.

C. Provide a backup to the hydrogen recombiners to prevent the concentration of hydrogen following an accident by the use of a containment purge system.

D. Purge of the containment during plant shutdown by supplying filtered and heated or cooled outside air through the high volume purge supply unit and exhausting the containment air through the high volume exhausting unit.

E. Purge the containment during plant operations by supplying filtered and heated or cooled outside air through the low volume purge supply unit and exhausting the containment air through a filtration unit.

The nuclear service water pumphouse is ventilated to maintain temperatures compatible with normal operating requirements.

### 1.2.9 RADIOACTIVE WASTE MANAGEMENT

The radioactive waste treatment systems provide all equipment necessary to collect, process, monitor, and dispose of radioactive liquid, gaseous, and solid wastes that are produced during reactor operation.

Liquid wastes containing potentially radioactive material are collected and monitored. Prior to discharge, equipment is provided for filtering, evaporating, and demineralizing the liquid as required. The treated water from the filters or demineralizers or the evaporator distillate may be recycled for use in the plant or may be discharged if it is within the limits of 10 CFR 20 and 10 CFR 50, Appendix I. A steam generator blowdown treatment system and a condensate polisher system are provided to permit continued plant operation with limited fuel clad defects concurrent with steam generator tube leaks.

Liquid and solid waste which cannot be either recycled in the plant or discharged to the environment are processed by the solid radwaste system for offsite disposal. Processing includes solidification of liquid wastes and packaging of all wastes into approved shipping containers.

Gaseous wastes are held up for radioactive decay in charcoal decay beds. Discharge to the environment is controlled to keep the offsite dose within the limits of 10 CFR 20 and 10 CFR 50, and Appendix I.

Table 1.2-1

1981 POPULATION BY SECTOR AND DISTANCE
WITHIN 16 KILOMETERS OF THE SITE

| Sector | 0-1.6 (km) | 1.6-3.2 (km) | 3.2-4.8 (km) | 4.8-6.4 (km) | 6.4-8.0 (km) | 8.0-16 (km) | Total 0 - 16 (km) |
|---|---|---|---|---|---|---|---|
| W | 0 | 1,218 | 2,738 | 236 | 415 | 4,940 | 9,547 |
| WNW | 0 | 2,649 | 110 | 170 | 171 | 5,706 | 8,806 |
| NW | 4,403 | 4,093 | 707 | 693 | 179 | 6,771 | 16,846 |
| NNW | 287 | 617 | 268 | 411 | 545 | 1,877 | 4,005 |
| N | 0 | 1,478 | 836 | 268 | 74 | 9,818 | 12,494 |
| NNE | 268 | 986 | 452 | 388 | 0 | 14,893 | 16,987 |
| NE | 0 | 1,097 | 2,420 | 1,018 | 952 | | 5,467 |
| ENE | 0 | 0 | 0 | 0 | 0 | 0 | 0 |
| E | 0 | 0 | 0 | 0 | 0 | 0 | 0 |
| ESE | 0 | 0 | 0 | 0 | 0 | 0 | 0 |
| SE | 0 | 0 | 0 | 0 | 0 | 0 | 0 |
| SSE | 0 | 0 | 0 | 0 | 0 | 0 | 0 |
| S | 0 | 0 | 0 | 0 | 0 | 0 | 0 |
| SSW | 0 | 0 | 0 | 0 | 334 | 12,022 | 12,356 |
| SW | 0 | 0 | 3,166 | 87 | 4,448 | 27,471 | 35,172 |
| WSW | 0 | 0 | 2,787 | 1,167 | 74 | 2,537 | 6,565 |
| Total | 4,958 | 12,118 | 13,484 | 4,438 | 7,192 | 86,035 | 128,225 |

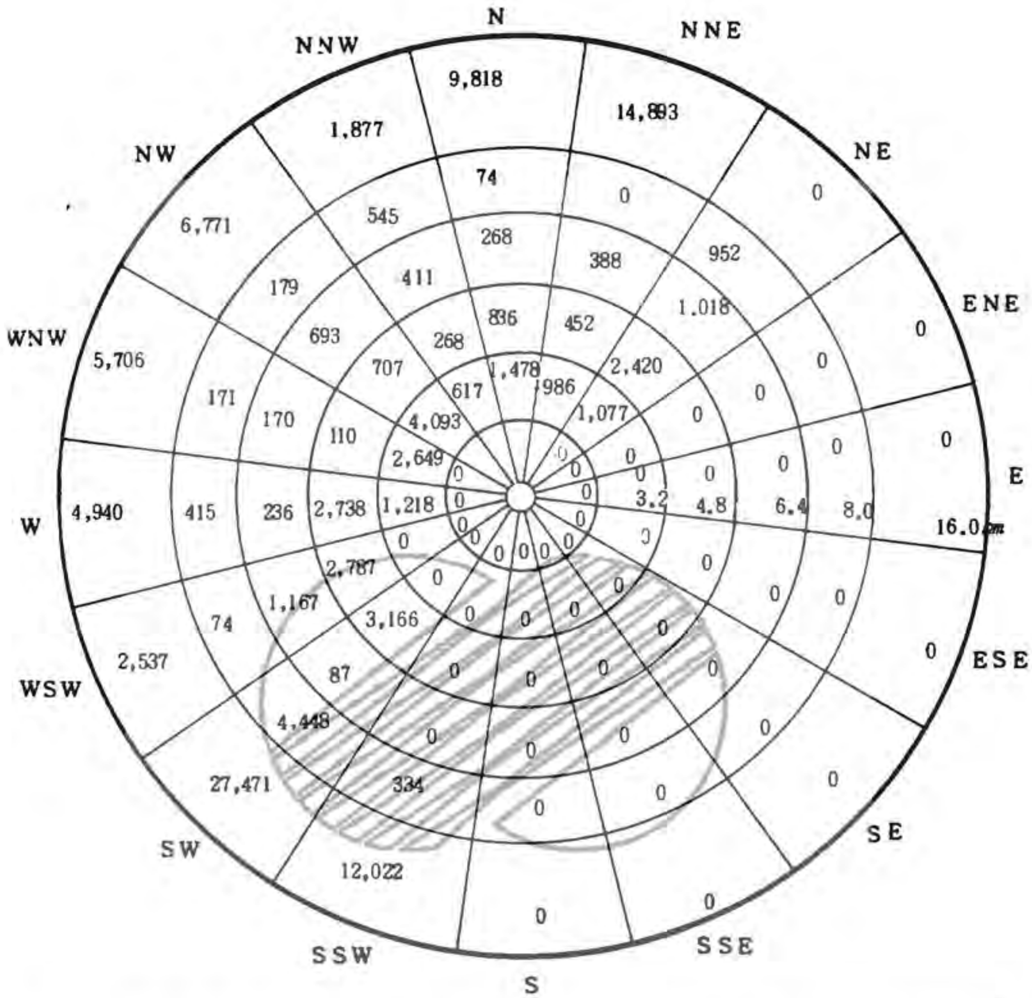
Table 1.2-2

1981 POPULATION BY SECTOR AND DISTANCE
BETWEEN 16 AND 80 KILOMETERS FROM THE SITE

(Population in thousands)

| Sector | 16 - 32 (km) | 32 - 48 (km) | 48 - 64 (km) | 64 - 80 (km) | Total 16 - 80 (km) |
|---|---|---|---|---|---|
| W | 41.0 | 46.9 | 182.7 | 277.6 | 548.2 |
| WNW | 13.5 | 31.3 | 53.9 | 73.6 | 172.3 |
| NW | 16.5 | 10.4 | 47.1 | 149.8 | 223.8 |
| NNW | 23.7 | 19.9 | 32.9 | 132.9 | 209.4 |
| N | 183.9 | 36.9 | 141.0 | 321.8 | 683.6 |
| NNE | 228.2 | 22.9 | 24.2 | 41.6 | 316.9 |
| NE | 0 | 0 | 0 | 0 | 0 |
| ENE | 0 | 0 | 0 | 0 | 0 |
| E | 0 | 0 | 0 | 0 | 0 |
| ESE | 0 | 0 | 0 | 0 | 0 |
| SE | 0 | 0 | 0 | 0 | 0 |
| SSE | 0 | 0 | 0 | 0 | 0 |
| S | 0 | 0 | 0 | 0 | 0 |
| SSW | 35.3 | 0 | 0 | 0 | 35.3 |
| SW | 1,660.5 | 889.4 | 4.4 | 70.3 | 2,624.6 |
| WSW | 474.3 | 206.8 | 64.5 | 100.9 | 846.5 |
| Total | 2,676.9 | 1,264.5 | 550.7 | 1,168.5 | 5,660.6 |

| Population within 1.6 km of the site | | | | |
|---|---|---|---|---|
| Direction | NW | NNW | N | NNE |
| Population | 4,403 | 287 | 0 | 268 |




| 0-1.6 (km) | 1.6-3.2 (km) | 3.2-4.8 (km) | 4.8-6.4 (km) | 6.4-8.0 (km) | 8.0-16 (km) | Total |
|---|---|---|---|---|---|---|
| 4,958 | 12,118 | 13,484 | 4,438 | 7,192 | 86,035 | 128,225 |

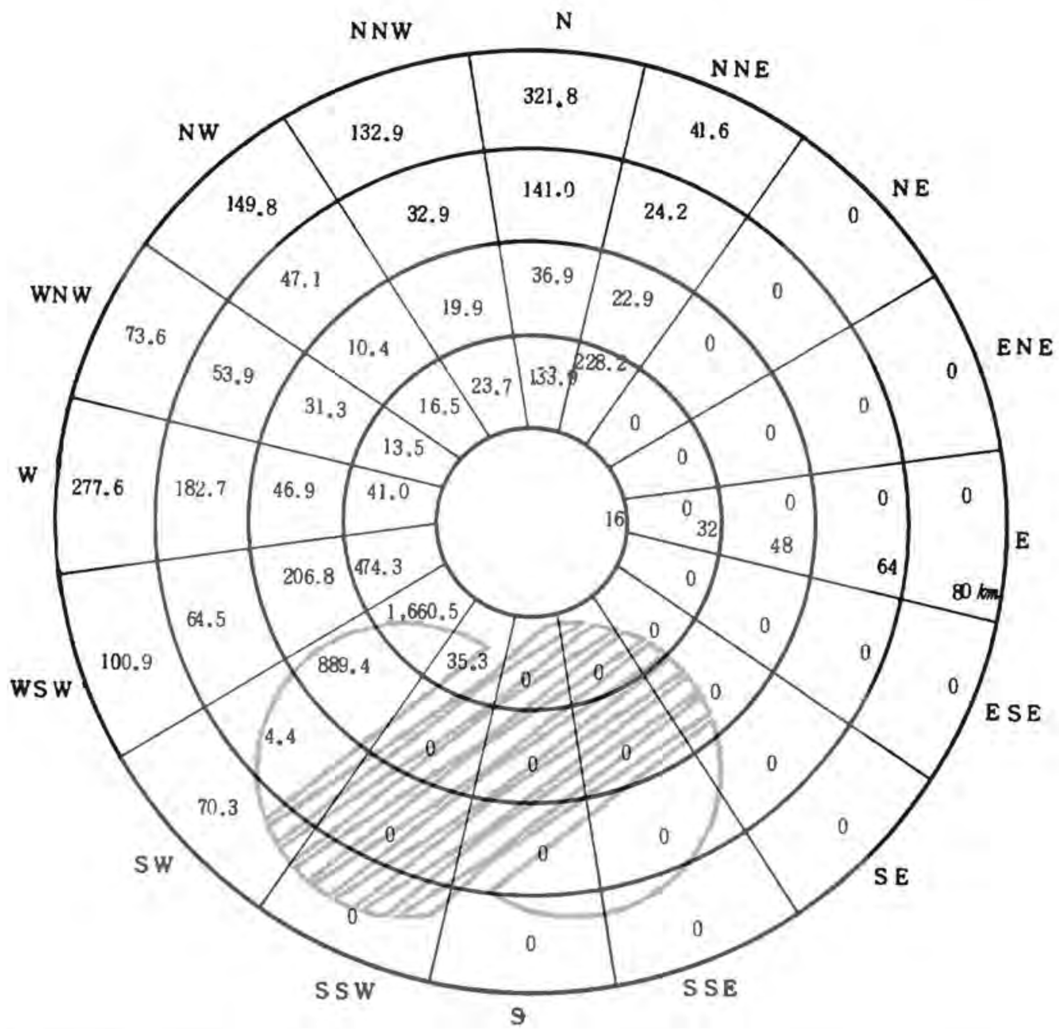
KOREA ELECTRIC POWER CORPORATION
KOREA NUCLEAR UNITS 5 & 6
FSAR

POPULATION DISTRIBUTION
WITHIN 10 MILES

Figure 1.2-1

2/17/83

(Population in thousands)




| 16 (km) 32 | 32 (km) 48 | 48 (km) 64 | 64 (km) 80 | Total |
|---|---|---|---|---|
| 2,676.9 | 1,264.5 | 550.7 | 1,168.5 | 5,660.6 |

2/17/83

KOREA ELECTRIC POWER CORPORATION
KOREA NUCLEAR UNITS 5 & 6
FSAR

POPULATION DISTRIBUTION
BETWEEN 10 AND 50 MILES

Figure 1.2-2

THIS FIGURE APPEARS
IN A SUPPLEMENT
TO THE FSAR

KOREA ELECTRIC POWER CORPORATION
KOREA NUCLEAR UNITS 5 & 6
FSAR

LOW POPULATION ZONE AND
EXCLUSION AREA BOUNDARY

Figure 1.2-3

THIS FIGURE APPEARS
IN A SUPPLEMENT
TO THE FSAR

KOREA ELECTRIC POWER CORPORATION
KOREA NUCLEAR UNITS 5 & 6
FSAR

EXCLUSION AREA BOUNDARY

Figure 1.2-4

THIS FIGURE APPEARS
IN A SUPPLEMENT
TO THE FSAR

KOREA ELECTRIC POWER CORPORATION
KOREA NUCLEAR UNITS 5 & 6
FSAR

SITE PLAN

Figure 1.2-5

KOREA ELECTRIC POWER CORPORATION
KOREA NUCLEAR UNITS 5 & 6
FSAR

GENERAL ARRANGEMENT
MISCELLANEOUS PLANS

Figure 1.2-6

KOREA HYDRO & NUCLEAR POWER COMPANY
KRN 3 & 4 FSAR

GENERAL ARRANGEMENT
PLAN EL. 59'-0" TO GRADE

Figure 1.2-7

KOREA ELECTRIC POWER CORPORATION
KOREA NUCLEAR UNITS 5 & 6
FSAR

GENERAL ARRANGEMENT
PLAN GRADE TO EL. 135'-6"

Figure 1.2-8

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KOREA HYDRO & NUCLEAR POWER COMPANY
KRN 3 & 4 FSAR

GENERAL ARRANGEMENT
PLAN AT TURBINE
OPERATING FLOOR EL. 135'-6"

Figure 1.2-9

KOREA ELECTRIC POWER CORPORATION
KOREA NUCLEAR UNITS 5 & 6
FSAR

GENERAL ARRANGEMENT
PLAN AT CONTAINMENT
OPERATING FLOOR EL. 148'-0"

Figure 1.2-10

KOREA HYDRO & NUCLEAR POWER COMPANY
KRN 3 & 4 FSAR

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GENERAL ARRANGEMENT
SECTION A

Figure 1.2-11

KOREA ELECTRIC POWER CORPORATION
KOREA NUCLEAR UNITS 5 & 6
FSAR

GENERAL ARRANGEMENT
SECTION B

Figure 1.2-12

KOREA ELECTRIC POWER CORPORATION
KOREA NUCLEAR UNITS 5 & 6
FSAR

GENERAL ARRANGEMENT
SECTION C AND E

Figure 1.2-13

KOREA HYDRO & NUCLEAR POWER COMPANY
KRN 3 & 4 FSAR

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GENERAL ARRANGEMENT
SECTION F D G H & J

Figure 1.2-14

KOREA ELECTRIC POWER CORPORATION
KOREA NUCLEAR UNITS 5 & 6
FSAR

GENERAL ARRANGEMENT
RADWASTE BUILDING

Figure 1.2-15

KOREA ELECTRIC POWER CORPORATION
KOREA NUCLEAR UNITS 5 & 6
FSAR

GENERAL ARRANGEMENT
EQUIPMENT LIST
(Sheet 1 of 3)

Figure 1.2-16

KOREA ELECTRIC POWER CORPORATION
KOREA NUCLEAR UNITS 5 & 6
FSAR

GENERAL ARRANGEMENT
EQUIPMENT LIST
(Sheet 2 of 3)

Figure 1.2-16

KOREA HYDRO & NUCLEAR POWER COMPANY
KRN 3 & 4 FSAR

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GENERAL ARRANGEMENT
EQUIPMENT LIST
(Sheet 3 of 3)

Figure 1.2-16

KOREA ELECTRIC POWER CORPORATION
KOREA NUCLEAR UNITS 5 & 6
FSAR

REACTOR COOLANT SYSTEM

Figure 1.2-17

KOREA HYDRO & NUCLEAR POWER COMPANY KORI 3 & 4 FSAR

AAC D/G BUILDING SITE PLAN

Figure 1 . 2-18

Amendment 533
2015.09.23

KOREA HYDRO & NUCLEAR POWER COMPANY
KRN 3&4 FSAR

SITE DEVELOPMENT
SEASHORE BARRIER WALL PLAN
FIGURE 1.2-19

## 1.3 COMPARISON TABLES

### 1.3.1 COMPARISON WITH SIMILAR FACILITY DESIGNS

This subsection highlights the principal differences and similarities between the major design features of this facility and other similar plants. In tables 1.3-1 and 1.3-2, comparisons with the other plants of similar design are listed. Table 1.3-1 documents the major similarities and differences of the nuclear steam supply system (NSSS) components and equipment of the facility with those of Maanshan (Taiwan Power Company) and Shearon Harris (Carolina Power and Light, North Carolina, USA) Nuclear Power Plants. In table 1.3-2, Maanshan, Pebble Springs, and SNUPPS have been listed for comparison with the facility concerning plant features other than the NSSS.

### 1.3.2 COMPARISON OF FINAL AND PRELIMINARY DESIGNS (FSAR)

Table 1.3-3 provides a listing of significant differences between the final design and preliminary design of the facility which have occurred since the submittal of the PSAR.

Table 1.3-1

NSSS VENDOR SUPPLIED COMPONENTS COMPARISON WITH SIMILAR FACILITY DESIGNS

(Sheet 1 of 33)

| Chapter | System/parameter | KRN 3 & 4 | Maanshan | Shearon Harris |
|---|---|---|---|---|
| 4 | Core Mechanical Design | Section 4.2 | | |
| | Fuel assemblies | | | |
| | Design | RCC canless | RCC canless | RCC canless |
| | Number of fuel assemblies | 157 | 157 | 157 |
| | $UO_2$ rods per assembly | 264 | 264 | 264 |
| | Rod pitch, mm. | 12.6 | 12.6 | 12.6 |
| | Overall dimensions, mm. | 214.02 × 214.02 | 215.04 × 215.04 | 215.04 × 215.04 376 |
| | Fuel weight (as $UO_2$), per assembly, kg | ≃ 521.7 | ≃ 476.3 | ≃ 498.92 |
| | Clad weight per assembly, kg | ≃ 107 | ≃ 118.8 | ≃ 128.7 273 |
| | Number of grids per assembly | 2 End(Top, Bottom) grid : Inconel 718<br>8 Mid grid : ZIRLO<br>3 IFM grid (RFA) : ZIRLO<br>5 IFM grid (ACF) : ZIRLO<br>1 Protective grid : Inconel 718 | 2 - Type R<br>6 - Type 2 | 8 - Type R 376 |
| | Loading technique | 3-region nonuniform | 3-region nonuniform | 3-region nonuniform |
| | Fuel rods | | | |
| | Number | 41,448 | 41,448 | 41,448 |
| | Outside diameter, mm. | 9.5 | 9.14 | 9.5 |

• RFA : Robust Fuel Assembly

Table 1.3-1

NSSS VENDOR SUPPLIED COMPONENTS COMPARISON WITH SIMILAR FACILITY DESIGNS
(Sheet 2 of 33)

| Chapter | System/Parameter | KRN 3 & 4 | Maanshan | Shearon Harris | |
|---|---|---|---|---|---|
| 4 | Diametral gap, mm. | 0.17 | 0.157 | 0.17 | 19 |
| | Clad thickness, mm. | 0.57 | 0.64 | 0.64 | |
| | Clad material | ZIRLO | Zircaloy-4 | Zircaloy-4 | 276 321 |
| | Fuel pellets | | | | |
| | Material | $UO_2/UO_2$-$Gd_2O_3$ sintered | $UO_2$ sintered | $UO_2$ sintered | |
| | Density, % of theoretical | 95 | 95 | 95 | 19 |
| | Diameter, mm. | 8.19 | 7.84 | 8.05 | 376 |
| | Length, mm. | 9.83 | 12.88 | 10.00 | 276 |
| | Rod cluster control assemblies | | | | |
| | Neutron absorber | Ag-In-Cd (full length) | $B_4C$ (with Hf or Ag-In-Cd Tips) | Ag-In-Cd or Hf | 19 302 |
| | Cladding material | Type 304(or 316L) SS-cold worked | Type 304 SS-cold worked | Type 304 SS-cold worked | |
| | Clad thickness, mm. | 0.47 | 0.57 | 0.47 | 19 |
| | Number of cluster full and part length | 52/0 | 52/0 | 52/0 | |

Table 1.3-1

NSSS VENDOR SUPPLIED COMPONENTS COMPARISON WITH SIMILAR FACILITY DESIGNS
(Sheet 3 of 33)

| Chapter | System/Parameter | KRN 3 & 4 | Maanshan | Shearon Harris |
|---|---|---|---|---|
| 4 | Number of absorber rods per cluster | 24 | 24 | 24 |
| | Core structure | | | |
| | Core barrel, ID/OD, cm. | 339.98 / 350.2 | 339.98 / 350.2 | 340.1/340.1 |
| | Thermal shield, ID/OD, cm. | Neutron pad design | Neutron pad design | Neutron pad design |
| | Structural characteristics | | | |
| | Core diameter, cm. (equivalent) | 304 | 304 | 304 |
| | Core height, cm. (active fuel) | 365.8 | 365.8 | 365.8 |
| | Reflector thickness and composition | | | |
| | Top - water plus steel, cm. | 25.4 | 25.4 | 25.4 |
| | Bottom - water plus steel, cm. | 25.4 | 25.4 | 25.4 |
| | Side - water plus steel, cm. | 38.1 | 38.1 | 38.1 |

Table 1.3-1

NSSS VENDOR SUPPLIED COMPONENTS COMPARISON WITH SIMILAR FACILITY DESIGNS
(Sheet 4 of 33)

| Chapter | System/Parameter | KRN 3 & 4 | Maanshan | Shearon Harris |
|---|---|---|---|---|
| 4 | $H_2O$/U molecular ratio core, lattice (cold) | 2.54 | 2.73 | 2.43 |
| | Feed enrichment, w/o | | | |
| | Region 1 | 1.60 | 1.60 | 2.10 |
| | Region 2 | 2.40 | 2.40 | 2.60 |
| | Region 3 | 3.10 | 3.10 | 3.10 |
| | Thermal and hydraulic design | Section 4.4 | | |
| | Reactor core heat output, MWt | 2,775 | 2,775 | 2,775 |
| | Reactor core heat output, $10^6$ Btu/h | 9,471 | 9,471 | 9,471 |
| | Heat generated in fuel, % | 97.4 | 97.4 | 97.4 |
| | System pressure, nominal, kg/㎠ [bar] | 160.3[157.2](4) | 157.2 | 155.1 |
| | System pressure, minimum steady-state, kg/㎠ [bar] | 158.2[155.1](4) | 155.1 | 153.1 |
| | Minimum DNBR for design transients | | $\geq$1.49 | $\geq$1.30 |
| | Typical Flow Channel | 1.35 | | |
| | Thimble Flow Channel | 1.33 | | |

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Table 1.3-1

NSSS VENDOR SUPPLIED COMPONENTS COMPARISON WITH SIMILAR FACILITY DESIGNS
(Sheet 4 of 33)

| Chapter | System/Parameter | KRN 3 & 4 | Maanshan | Shearon Harris |
|---|---|---|---|---|
| 4 | $H_2O/U$ molecular ratio core, lattice (cold) | 2.54 | 2.73 | 2.43 |
| | Feed enrichment, w/o | | | |
| | Region 1 | 1.60 | 1.60 | 2.10 |
| | Region 2 | 2.40 | 2.40 | 2.60 |
| | Region 3 | 3.10 | 3.10 | 3.10 |
| | Thermal and hydraulic design | Section 4.4 | | |
| | Reactor core heat output, MWt | 2,900 | 2,775 | 2,775 |
| | Reactor core heat output, $10^6$ Btu/h | 9,898 | 9,471 | 9,471 |
| | Heat generated in fuel, % | 97.4 | 97.4 | 97.4 |
| | System pressure, nominal, kg/cm² [bar] | 158.2[155.1] | 157.2 | 155.1 |
| | System pressure, minimum steady-state, kg/cm² [bar] | 154.3[151.3] | 155.1 | 153.1 |
| | Minimum DNBR for design transients | | ≥1.49 | ≥1.30 |
| | Typical Flow Channel | 1.25 | | |
| | Thimble Flow Channel | 1.25 | | |

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Table 1.3-1

NSSS VENDOR SUPPLIED COMPONENTS COMPARISON WITH SIMILAR FACILITY DESIGNS (17×17 ACE7)

(Sheet 4 of 33)

| Chapter | System/parameter | KRN 3 & 4 | Maanshan | Shearon Harris |
|---|---|---|---|---|
| 4 | $H_2O/U$ molecular ratio core, lattice (cold) | 2.54 | 2.73 | 2.43 |
| | Feed enrichment, w/o | | | |
| | Region 1 | 1.60 | 1.60 | 2.10 |
| | Region 2 | 2.40 | 2.40 | 2.60 |
| | Region 3 | 3.10 | 3.10 | 3.10 |
| | Thermal and hydraulic design | Section 4.4 | | |
| | Reactor core heat output, MWt | 2,900 | 2,775 | 2,775 |
| | Reactor core heat output, $10^6$ Btu/h | 9,898 | 9,471 | 9,471 |
| | Heat, generated, in fuel, % | 97.4 | 97.4 | 97.4 |
| | System pressure, nominal, bar | 155.1 | 157.2 | 155.1 |
| | System pressure, minimum steady-state, bar | 151.3 | 155.1 | 153.1 |
| | Minimum DNBR for design transients | | ≥ 1.49 | ≥ 1.30 |
| | Typical Flow Channel | **1.23** | | |
| | Thimble Flow Channel | **1.23** | | |

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Table 1.3-1

NSSS VENDOR SUPPLIED COMPONENTS COMPARISON WITH SIMILAR FACILITY DESIGNS
(Sheet 5 of 33)

| Chapter | System/Parameter | KRN 3 & 4 | Maanshan | Shearon Harris |
|---|---|---|---|---|
| 4 | Coolant flow | | | |
| | Total thermal flow rate, kg/sec(5) | 13,331 | 14,099 | 13,620 |
| | Effective flow rate for heat transfer, kg/sec | 12,436 | 13,444 | 13,032 |
| | Effective flow area for heat transfer, $m^2$ | 3.86 | 4.09 | 3.87 |
| | Average velocity along fuel rods, m/sec | 4.30 | 4.66 | 4.69 |
| | Average mass velocity, $kg/m^2 \cdot sec$ | 3,221 | 3,287 | 3,380 |
| | Coolant temperature, ℃ | | | |
| | Nominal inlet | 289.4 | 292.2 | 291.1 |
| | Average rise in vessel | 37.8 | 34.1 | 35.4 |
| | Average rise in core | 40.1 | 35.9 | 37.0 |
| | Average in core | 310.7 | 310.9 | 310.7 |
| | Average in vessel | 308.3 | 310.1 | 314.6 |

* V5H : Vantage - 5H
RFA : Robust Fuel assembly

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Table 1.3-1a

NSSS VENDOR SUPPLIED COMPONENTS COMPARISON WITH SIMILAR FACILITY DESIGNS (17×17 ACE7)

(Sheet 5 of 33)

| Chapter | System/Parameter | KRN 3 & 4 | Maanshan | Shearon Harris |
|---|---|---|---|---|
| 4 | Coolant flow | | | |
| | Total thermal flow rate, kg/sec(5) | 13,192 | 14,099 | 13,620 |
| | Effective flow rate for heat transfer, kg/sec | 12,058 | 13,444 | 13,032 |
| | Effective flow area for heat transfer, $m^2$ | 3.86 | 4.09 | 3.87 |
| | Average velocity along fuel rods, m/sec | 4.18 | 4.66 | 4.69 |
| | Average mass velocity, $kg/m^2 \cdot sec$ | 3.123 | 3.287 | 3.380 |
| | Coolant temperature, ℃ | | | |
| | Nominal inlet | **289.2** | 292.2 | 291.1 |
| | Average rise in vessel | **38.1** | 34.1 | 35.4 |
| | Average rise in core | **41.2** | 35.9 | 37.0 |
| | Average in core | **311.2** | 310.9 | 310.7 |
| | Average in vessel | **308.3** | 310.1 | 314.6 |

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Table 1.3-1

NSSS VENDOR SUPPLIED COMPONENTS COMPARISON WITH SIMILAR FACILITY DESIGNS
(Sheet 6 of 33)

| Chapter | System/Parameter | KRN 3 & 4 | Maanshan | Shearon Harris |
|---|---|---|---|---|
| 4 | Heat transfer | | | |
| | Active heat transfer surface area, $m^2$ | 4,550 | 4,350 | 4,550 |
| | Average heat flux, $kW/m^2$ | 626 | 638 | 610 |
| | Maximum heat flux for normal operation, $kW/m^2$ | 1,627 | 1,481 | 1,415 |
| | Average linear power, kW/ft | 5.69 | 5.44 | 5.44 |
| | Peak linear power for normal operation, kW/ft | 14.8 | 12.6 | 12.6 |
| | Peak linear power resulting from over-power transients/operator error (assuming a maximum overpower of 118%) kW/ft | < 22.5 | 18.0 | 18.0 |
| | Heat flux hot channel factor, $F_Q$ | 2.60 | 2.30 | 2.32 |

(1) This limit is associated with the value of FQ=2.60
(2) See subparagraph 4.3.2.2.6
(3) This is the value of Fa for normal operation.
(4) Values used for thermal hydraulic analysis
(5) Based on the Thermal Design Flow Rate
* ( ) : Values for V-5H fuel

19 321 4 273 321 4

Table 1.3-1

NSSS VENDOR SUPPLIED COMPONENTS COMPARISON WITH SIMILAR FACILITY DESIGNS
(Sheet 7 of 33)

| Chapter | System/Parameter | KRN 3 & 4 | Maanshan | Shearon Harris |
|---|---|---|---|---|
| 4 | Peak fuel centerline temperature at peak linear power for prevention of centerline melt, °C | 2,593-3.75xWg(w/o $Gd_2O_3$) | 2,593 | 2,593 |
| 5 | RCS Design and Operating Parameters | Section 5.1 | | |
| | Plant design life, yr | 40 | 40 | 40 |
| | Nominal operating pressure, psig | 2,235 | 2,235 | 2,235 |
| | Total system volume including pressurizer and surge line, $ft^3$ | 9,275 | 9,410 | 9,723 |
| | System liquid volume, including pressurizer water at maximum guaranteed power, $ft^3$ | 8,700 | 8,833 | 8,963 |
| | Pressurizer spray rate, maximum gpm | 700 | 700 | 600 |
| | Pressurizer heater capacity, kW | 1,400 | 1,400 | 1,400 |
| | Pressurizer relief tank volume, $ft^3$ | 1,300 | 1,300 | 1,300 |

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Table 1.3-1

NSSS VENDOR/SUPPLIED COMPONENTS COMPARISON WITH SIMILAR FACILITY DESIGNS
(Sheet 8 of 33)

| Chapter | System/Parameter | KRN 3 & 4 | | | Maanshan | Shearon Harris |
|---|---|---|---|---|---|---|
| 5 | RCS thermal and hydraulic data | | | | | |
| | NSSS power, MWt | 2,912 | | | 2,785 | 2,785 |
| | Thermal design flows, gpm | | | | | |
| | Active loop | 94,200 | | | 97,600 | 97,600 |
| | Idle loop | - | | | - | - |
| | Reactor | 282,600 | | | 292,800 | 292,800 |
| | Tavg(°F) | 587 | 586 | 580 | | |
| | Total reactor flow, $10^6$ lb/h | 105.8 | 105.9 | 106.9 | 109.1 | 109.2 |
| | Temperatures, °F | | | | | |
| | Reactor vessel outlet | 621.0 | 620.0 | 614.4 | 619.9 | 619.0 |
| | Reactor vessel inlet | 553.0 | 552.0 | 545.6 | 557.0 | 556.0 |
| | steam generator outlet | 552.7 | 551.7 | 545.3 | 556.7 | 555.8 |
| | Steam generator steam | 535.4 | 534.3 | 527.7 | 540.2 | 540.2 |
| | Feedwater | 445.9 | 445.9 | 445.9 | 440.0 | 435.0 |

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Table 1.3-1

NSSS VENDOR SUPPLIED COMPONENTS COMPARISON WITH SIMILAR FACILITY DESIGNS
(Sheet 9 of 33)

| Chapter | System/Parameter | KRN 3 & 4 | | | Maanshan | Shearon Harris |
|---|---|---|---|---|---|---|
| 5 | Steam pressure, psia | 926.0 | 918.0 | 868.0 | 964 | 964 |
| | Total steam flow, $10^6$ lb/h | 12.94 | 12.93 | 12.91 | 12.3 | 12.2 |
| | Best estimate flows, gpm | | | | | |
| | Active loop | | 102,600 | | 102,800 | 103,000 |
| | Reactor | | 307,800 | | 308,400 | 309,000 |
| | Mechanical design flows, gpm | | | | | |
| | Active loop | | 106,900 | | 106,900 | 107,100 |
| | Reactor | | 320,700 | | 320,700 | 321,300 |
| | System pressure drops | | | | | |
| | Reactor vessel ΔP, psi | | 41.0 | | 41.0 | 43.8 |
| | Steam generator ΔP, psi | | 41.0 | | 41.0 | 38.7 |
| | Hot leg piping ΔP, psi | | 1.4 | | 1.4 | 2.0 |
| | Pump suction piping ΔP, psi | | 3.4 | | 3.4 | 3.6 |
| | Cold leg piping ΔP, psi | | 3.4 | | 3.4 | 2.0 |
| | Pump head, ft | | 280 | | 280 | 279 |

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Table 1.3-1

NSSS VENDOR SUPPLIED COMPONENTS COMPARISON WITH SIMILAR FACILITY DESIGNS
(Sheet 10 of 33)

| Chapter | System/Parameter | KNU 5 & 6 | Maanshan | Shearon Harris |
|---|---|---|---|---|
| 5 | RCS design pressure settings | Section 5.2 | | |
| | Hydrostatic test pressure, psig | 3,107 | 3,107 | 3,107 |
| | Design pressure, psig | 2,485 | 2,485 | 2,485 |
| | Safety valves (begin to open), psig | 2,485 | 2,485 | 2,485 |
| | High pressure reactor trip, psig | 2,385 | 2,385 | 2,385 |
| | High pressure alarm, psig | 2,310 | 2,310 | 2,335 |
| | Power relief valves, psig | 2,335(4) | 2,335 | 2,335 |
| | Pressurizer spray valves (full open), psig | 2,310 | 2,310 | 2,310 |
| | Pressurizer spray valves (begin to open), psig | 2,260 | 2,260 | 2,260 |
| | Proportional heaters, (begin to operate), psig | 2,250 | 2,250 | 2,250 |
| | Operating pressure, psig | 2,235 | 2,235 | 2,235 |
| | Proportional heater (full operation), psig | 2,220 | 2,220 | 2,220 |

(4) At 2,335 psig, a pressure signal initiates actuating (opening) of these valves, remote manual control is also provided.

Table 1.3-1

NSSS VENDOR SUPPLIED COMPONENTS COMPARISON WITH SIMILAR FACILITY DESIGNS
(Sheet 11 of 33)

| Chapter | System/Parameter | KNU 5 & 6 | Maanshan | Shearon Harris |
|---|---|---|---|---|
| 5 | Backup heaters on, psig | 2,210 | 2,210 | 2,210 |
| | Pressurizer relief valve interlock, psig | 2,185(5) | 2,335 | 2,185 |
| | Low pressure reactor trip (typical, but variable), psig | 1,945 | 1,970 | 1,825 |
| | Reactor vessel design | | | |
| | Design/operating pressure, psig | 2485/2235 | 2,485/2,235 | 2,485/2,235 |
| | Design temperature, °F | 650 | 650 | 650 |
| | Overall height of vessel and closure head, ft-in. (bottom head outside diameter to top of control rod mechanism adapter) | 42-7 3/16 | 42-7 3/16 | 42-7 3/16 |
| | Thickness of insulation, minimum, in. | 3 | 3 | 3 |
| | Number of reactor closure head studs | 58 | 58 | 58 |

(5) This interlock closes the power operated relief valves and their block valves when pressure drops to indicated value.

Table 1.3-1

NSSS VENDOR SUPPLIED COMPONENTS COMPARISON WITH SIMILAR FACILITY DESIGNS
(Sheet 12 of 33)

| Chapter | System/Parameter | KNU 5 & 6 | Maanshan | Shearon Harris |
|---|---|---|---|---|
| 5 | Diameter of reactor closure head/ studs, in. (minimum shank) | 6 | 6 | 6 |
| | Inside diameter of flange, in. | 149 9/16 | 149 9/16 | 149 9/16 |
| | Outside diameter of flange, in. | 184 | 184 | 184 |
| | Inside diameter at shell, in. | 157 | 157 | 157 |
| | Inlet nozzle inside diameter, maximum, in. | 27 1/2 | 27 1/2 | 27 1/2 |
| | Outlet nozzle inside diameter, maximum, in. | 29 | 29 | 29 |
| | Clad thickness, minimum, in. | 1/8 | 1/8 | 1/8 |
| | Lower head thickness, minimum, in. | 5 | 5 | 5 |
| | Vessel belt-line thickness, minimum, in. | 7 7/8 | 7 7/8 | 7 7/8 |
| | Closure head thickness, minimum, in. | 6 3/16 | 6 3/16 | 6 3/16 |

Table 1.3-1

NSSS VENDOR SUPPLIED COMPONENTS COMPARISON WITH SIMILAR FACILITY DESIGNS
(Sheet 13 of 33)

| Chapter | System/Parameter | KRN 3 & 4 | Maanshan | Shearon Harris |
|---|---|---|---|---|
| 5 | Reactor coolant pump design | | | |
| | Unit design pressure, psig | 2,485 | 2,485 | 2,485 |
| | Unit design temperature, °F | 650[a] | 650 | 650 |
| | Unit overall height, ft | 27.5 | 26.93 | 26 |
| | Seal water injection, gpm | 8 | 8 | 8 |
| | Seal water return, gpm | 3 | 3 | 3 |
| | Cooling water flow, gpm per RCP | 516 | 596 | 195 |
| | Maximum continuous cooling water inlet temperature, °F | 105 | 105 | 105 |
| | Pump | | | |
| | suction temperature, °F | 556.7 | 557 | |
| | Discharge nozzle, ID, in. | 27-1/2 | 27-1/2 | 27-1/2 |
| | Suction nozzle, ID, in. | 31 | 31 | 31 |
| | Speed, rpm | 1,188 | 1,185 - 1,190 | 1,183 |

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(6) Design temperature of pressure retaining parts of the pump assembly exposed to the reactor coolant and injection water on the high pressure side of the controlled leakage seal shall be that temperature determined for the parts for a primary loop temperature of 650 °F.

Table 1.3-1

NSSS VENDOR SUPPLIED COMPONENTS COMPARISON WITH SIMILAR FACILITY DESIGNS
(Sheet 14 of 33)

| Chapter | System/parameter | KRN 3 & 4 | Maanshan | Shearon Harris |
|---|---|---|---|---|
| 5 | Water volume, $ft^3$ | 80(7) | 80 | 57 |
| | Weight (dry), lb | 207,000 | 203,500 | 201,900 |
| | Motor | | | |
| | Type | Drip-proof squirrel cage induction water/air cooled, totally enclosed | Drip-proof, squirrel cage | Drip-proof, squirrel cage induction air cooled |
| | Power, hp | 7,000 | 7,000 | 7,000 |
| | Voltage, volts | 13,200 | 13,200 | 6,600 |
| | Phase | 3 | 3 | 3 |
| | Frequency, Hz | 60 | 60 | 60 |
| | Current | | | |
| | starting | 1900 amp @ 13,200 volts | 1750 amp @ 13,200 volts | 3,000 amp @ 6,600 volts |
| | Input, hot reactor coolant | 271 amp | 254 ± 5 amp | ~491 amp |
| | Input, cold reactor coolant | 338 amp | 336 ± 7 amp | ~660 amp |

(7) Composed of reactor coolant in casing and of injection and cooling water in the thermal barrier.

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Table 1.3-1

NSSS VENDOR SUPPLIED COMPONENTS COMPARISON WITH SIMILAR FACILITY DESIGNS
(Sheet 15 of 33)

| Chapter | System/Parameter | KNU 5 & 6 | Maanshan | Shearon Harris |
|---|---|---|---|---|
| 5 | Pump moment of inertia, lb-$ft^2$ maximum | | | |
| | Flywheel | 70,000 | 70,000 | 70,000 |
| | Motor | 22,500 | 22,500 | 22,500 |
| | Shaft | 520 | 520 | 520 |
| | Impeller | 1980 | 1,980 | 1,980 |
| | Steam generator design | | | |
| | Design pressure, reactor coolant side, psig | 2,485 | 2,485 | 2,485 |
| | Design pressure, steam side, psig | 1,185 | 1,185 | 1,185 |
| | Design temperature, reactor coolant side, °F | 650 | 650 | 650 |
| | Design temperature, steam side, °F | 600 | 600 | 600 |
| | Total heat transfer surface area, $ft^2$ | 55,000 | 55,000 | 48,000 |
| | Maximum moisture carryover, wt% | 0.25 | 0.25 | 0.25 |
| | Overall height, ft-in. | 67-8 | 67-8 | 67-8 |
| | Number of U-tubes | 5,626 | 5,626 | 4,674 |

Table 1.3-1

NSSS VENDOR SUPPLIED COMPONENTS COMPARISON WITH SIMILAR FACILITY DESIGNS
(Sheet 16 of 33)

| Chapter | System/Parameter | KNU 5 & 6 | Maanshan | Shearon Harris |
|---|---|---|---|---|
| 5 | U-Tube outer diameter, in. | 0.688 | 0.688 | 0.750 |
| | Tube wall thickness, nominal, in. | 0.040 | 0.040 | 0.043 |
| | Number of manways | 4 | 4 | 4 |
| | Inside diameter of manways, in. | 16 | 16 | 16 |
| | Design fouling factor | 0.00006 | 0.00006 | 0.00005 |
| | Reactor coolant piping design | | | |
| | Reactor inlet piping, inside diameter, in. | 27-1/2 | 27-1/2 | 27-1/2 |
| | Reactor inlet piping, nominal wall thickness, in. | 2.32 | 2.32 | 2.32 |
| | Reactor outlet piping, inside diameter, in. | 29 | 29 | 29 |
| | Reactor outlet piping, nominal wall thickness, in. | 2.45 | 2.45 | 2.45 |

Table 1.3-1

NSSS VENDOR SUPPLIED COMPONENTS COMPARISON WITH SIMILAR FACILITY DESIGNS
(Sheet 17 of 33)

| Chapter | System/Parameter | KNU 5 & 6 | Maanshan | Shearon Harris |
|---|---|---|---|---|
| 5 | Coolant pump suction piping, inside diameter, in. | 31 | 31 | 31 |
| | Coolant pump suction piping, nominal wall thickness, in. | 2.60 | 2.60 | 2.60 |
| | Pressurizer surge line piping, nominal pipe size, in. | 14 | 14 | 14 |
| | Pressurizer surge line piping, nominal wall thickness, in. | 1.406 | 1.406 | 1.406 |
| | Reactor coolant loop piping | | | |
| | Design/operating pressure, psig | 2485/2235 | 2,485/ 2,235 | 2,485/2,235 |
| | Design temperature, °F | 650 | 650 | 650 |
| | Pressurizer surge line | | | |
| | Design pressure, psig | 2,485 | 2,485 | 2,485 |
| | Design temperature, °F | 680 | 680 | 680 |

Table 1.3-1

NSSS VENDOR SUPPLIED COMPONENTS COMPARISON WITH SIMILAR FACILITY DESIGNS
(Sheet 18 of 33)

| Chapter | System/Parameter | KNU 5 & 6 | Maanshan | Shearon Harris |
|---|---|---|---|---|
| 5 | Pressurizer safety valve inlet line | | | |
| | Design pressure, psig | 2,485 | 2,485 | 2,485 |
| | Design temperature, °F | 680 | 680 | 680 |
| | Pressurizer power-operated relief valve inlet line | | | |
| | Design pressure, psig | 2,485 | 2,485 | 2,485 |
| | Design temperature, °F | 680 | 680 | 680 |
| | Pressurizer relief tank inlet line | | | |
| | Design pressure, psig | 600 | 600 | 600 |
| | Design temperature °F | 600 | 600 | 600 |
| | RHRS Design bases | Subsection 5.4.7 | | |
| | Residual heat removal system startup | ~4 hours after reactor shutdown | ~4 hours after reactor shutdown | ~4 hours after reactor shutdown |
| | Reactor coolant system initial pressure, psig | ~400 | ~425 | 400 |

Table 1.3-1

NSSS VENDOR SUPPLIED COMPONENTS COMPARISON WITH SIMILAR FACILITY DESIGNS
(Sheet 19 of 33)

| Chapter | System/parameter | KRN 3 & 4 | Maanshan | Shearon Harris | |
|---|---|---|---|---|---|
| 5 | Reactor coolant system initial temperature, °F | ~350 | ~350 | 350 | |
| | Component cooling water temperature, °F | 105 | 105 | 105 | 321 |
| | Cooldown time, hours after initiation of residual heat removal system operation | ~16 | ~16 | 16 | |
| | Reactor coolant system temperature at end of cooldown, °F | 140 | 140 | 140 | |
| | Decay heat generation at 20 hours after reactor shutdown, Btu/h | $64.2 \times 10^6$ | $62.4 \times 10^6$ | $59 \times 10^6$ | 321 |
| | Residual heat removal pump | | | | |
| | Number | 2 | 2 | 2 | |
| | Design pressure, psig | 600 | 600 | 600 | |
| | Design temperature, °F | 400 | 400 | 400 | |

Table 1.3-1

NSSS VENDOR SUPPLIED COMPONENTS COMPARISON WITH SIMILAR FACILITY DESIGN
(Sheet 20 of 33)

| Chapter | System/Parameter | KNU 5 & 6 | Maanshan | Shearon Harris |
|---|---|---|---|---|
| 5 | Design flow, gpm | 3,000 | 3,000 | 3,750 |
| | Design head, ft | 270 | 270 | 240 |
| | Residual heat exchanger | | | |
| | Number | 2 | 2 | 2 |
| | Design heat removal capacity, Btu/h | $31.2 \times 10^6$ | $31.2 \times 10^6$ | $29.5 \times 10^6$ |

Table 1.3-1

NSSS VENDOR SUPPLIED COMPONENTS COMPARISON WITH SIMILAR FACILITY DESIGNS
(Sheet 21 of 33)

| Chapter | System/Parameter | KNU 5 & 6 | | Maanshan | | Shearon Harris | |
|---|---|---|---|---|---|---|---|
| | | Tube-Side | Shell-Side | Tube-Side | Shell-Side | Tube-Side | Shell-Side |
| | RHR heat exchanger (cont'd) | | | | | | |
| 5 | Design pressure, psig | 600 | 150 | 600 | 150 | 600 | 150 |
| | Design temp, °F | 400 | 200 | 400 | 200 | 400 | 200 |
| | Design flow, lb/h | $1.5 \times 10^6$ | $2.6 \times 10^6$ | $1.5 \times 10^6$ | $2.6 \times 10^6$ | $1.87 \times 10^6$ | $2.8 \times 10^6$ |
| | Inlet temp, °F | 140 | 105 | 140 | 105 | 140 | 105 |
| | Outlet temp, °F | 119.2 | 117.2 | 119.2 | 117 | 124.3 | 116.5 |
| | Material | Austenitic stainless steel | Carbon steel | Austenitic stainless steel | Carbon steel | Austenitic stainless steel | Carbon steel |
| | Fluid | Reactor coolant | Component cooling water | Reactor coolant | Component cooling water | Reactor coolant | Component cooling water |

Table 1.3-1

NSSS VENDOR SUPPLIED COMPONENTS COMPARISON WITH SIMILAR FACILITY DESIGNS
(Sheet 22 of 33)

| Chapter | System/Parameter | KNU 5 & 6 | Maanshan | Shearon Harris |
|---|---|---|---|---|
| 5 | Pressurizer design | | | |
| | Design pressure, psig | 2,485 | 2,485 | 2,485 |
| | Design temperature, °F | 680 | 680 | 680 |
| | Surge line nozzle diameter, in. | 14 | 14 | 14 |
| | Internal volume, $ft^3$ | 1,400 | 1,400 | 1,400 |
| | Pressurizer relief tank design | | | |
| | Design pressure, psig | 100 | 100 | 100 |
| | Rupture disc release pressure, psig | Nominal: 91<br>Range: 86-100 | Nominal: 91<br>Range: 86-100 | 85 ± 5% |
| | Design temperature, °F | 340 | 340 | 340 |
| | Total rupture disc relief capacity, lb/h at 100 psig | $1.14 \times 10^6$ | $1.14 \times 10^6$ | $1.14 \times 10^6$ |

Table 1.3-1

NSSS VENDOR SUPPLIED COMPONENTS COMPARISON WITH SIMILAR FACILITY DESIGNS
(Sheet 23 of 33)

| Chapter | System/Parameter | KNU 5 & 6 | Maanshan | Shearon Harris |
|---|---|---|---|---|
| 5 | RCS valve design | | | |
| | Design/normal operating pressure, psig | 2,485/2,235 | 2,485/2,235 | 2,485/2,235 |
| | Preoperational plant hydrotest, psig | 3,107 | 3,107 | 3,107 |
| | Design temperature, °F | 650 | 650 | 650 |
| 6 | ECCS design | Section 6.3 | | |
| | Charging pumps | | | |
| | Pump design parameters | | | |
| | Number | 3 | 3 | 3 |
| | Type | Horizontal centrifugal | Horizontal centrifugal | Horizontal centrifugal |
| | Design pressure, psig | 2,800 | 2,800 | 2,800 |
| | Design temperature, °F | 300 | 300 | 300 |
| | Design flow rate, gpm | 150 | 150 | 150 |
| | Maximum flow rate, gpm | 650 | 650 | 650 |

Table 1.3-1

NSSS VENDOR SUPPLIED COMPONENTS COMPARISON WITH SIMILAR FACILITY DESIGNS
(Sheet 24 of 33)

| Chapter | System/Parameter | KNU 5 & 6 | Maanshan | Shearon Harris |
|---|---|---|---|---|
| 6 | Design head, ft | 5,800 | 5,800 | 5,800 |
| | Shutoff head, ft | 6,200 | 6,200 | 6,150 |
| | Residual heat removal pumps | | | |
| | Pump design parameters | | | |
| | Number | 2 | 2 | 2 |
| | Type | Vertical centrifugal | Vertical centrifugal | Vertical centrifugal |
| | Design pressure, psig | 600 | 600 | 600 |
| | Design temperature, °F | 400 | 400 | 400 |
| | Design flow, per pump, gpm | 3,000 | 3,000 | 3,750 |
| | Design head, at maximum flowrate ft | 270 | 270 | 240 |
| | Boron injection recirculation pumps | | | |
| | Pump design parameters | | | |
| | Number | 2 | 2 | 2 |
| | Type | Canned centrifugal | Canned centrifugal | Canned centrifugal |

Table 1.3-1

NSSS VENDOR SUPPLIED COMPONENTS COMPARISON WITH SIMILAR FACILITY DESIGNS
(Sheet 25 of 33)

| Chapter | System/Parameter | KNU 5 & 6 | Maanshan | Shearon Harris |
|---|---|---|---|---|
| 6 | Design pressure, psig | 150 | 150 | 240 |
| | Design temperature, °F | 250 | 250 | 200 |
| | Design flow, per pump, gpm | 20 | 20 | 20 |
| | Design head, ft | 100 | 100 | 100 |
| | Accumulator design parameters | | | |
| | Number | 3 | 3 | 3 |
| | Design pressure, psig | 700 | 700 | 700 |
| | Design temperature, °F | 300 | 300 | 300 |
| | Operating temperature, °F | 100-150 | 70-150 | 100-150 |
| | Normal operating pressure, psig | 660 | 660 | 660 |
| | Minimum operating pressure, psig | 600 | 600 | 600 |
| | Total volume, $ft^3$ | 1,450 each | 1,450 | 1,450 |

Table 1.3-1

NSSS VENDOR SUPPLIED COMPONENTS COMPARISON WITH SIMILAR FACILITY DESIGNS
(Sheet 26 of 33)

| Chapter | System/Parameter | KNU 5 & 6 | Maanshan | Shearon Harris |
|---|---|---|---|---|
| 6 | Minimum water volume at operating conditions, $ft^3$ (each) | 1,000 | 1,000 | 1,000 |
| | Nominal boron concentration, ppmB | 1,950 | 1,950 | 2,000 |
| | Boron injection tank design parameters | | | |
| | Number | 1 | 1 | 1 |
| | Total volume, gal | 900 | 900 | 900 |
| | Nominal boric acid concentration, ppmB | 21,000 | 21,000 | 21,000 |
| | Design pressure, psig | 2,735 | 2,735 | 2,735 |
| | Design temperature, °F | 300 | 300 | 300 |
| | Operating temperature, °F | 155-175 | 155-175 | 155-175 |
| | Heater banks, number | 2 | 2 | 2 |
| | Heater power, kW/heater bank | 6 | 6 | 6 |

Table 1.3-1

NSSS VENDOR SUPPLIED COMPONENTS COMPARISON WITH SIMILAR FACILITY DESIGNS
(Sheet 27 of 33)

| Chapter | System/Parameter | KNU 5 & 6 | Maanshan | Shearon Harris |
|---|---|---|---|---|
| 6 | Boron injection surge tank parameters | | | |
| | Number | 1 | 1 | 1 |
| | Total volume, gal | 75 | 75 | 75 |
| | Design pressure, psig | Atmos-pheric | Atmos-pheric | Atmos-pheric |
| | Operating pressure, psig | Atmos-pheric | Atmos-pheric | Atmos-pheric |
| | Design temperature, °F | 200 | 200 | 200 |
| | Operating temperature, °F | 155-175 | 155-175 | 155-175 |
| | Heater banks, number | 1 | 1 | 1 |
| | Heater bank power, kW | 6 | 6 | 6 |

Table 1.3-1
NSSS VENDOR SUPPLIED COMPONENTS COMPARISON WITH SIMILAR FACILITY DESIGNS
(Sheet 28 of 33)

| Chapter | Valves | | KNU 5 & 6 | Maanshan | Shearon Harris |
|---|---|---|---|---|---|
| 6 | 1. All motor-operated valves which must function on SIS | Maximum opening or closing time: | | | |
| | a. up to and including 8 in. | Time, s | 15 | 15 | 15 |
| | b. Over 8 in. | Valve stem travel, in./min | 49 | 49 | 49 |
| | 2. All other motor-operated valves | Maximum opening or closing time: | | | |
| | a. Up to and including 8 in. | Valve stem travel, in./min | 12 | 12 | 12 |
| | b. over 8 in. | Time, s | 120 | 120 | 120 |
| | 3. Leakage | | | | |
| | a. Conventional globe valves | Disc leakage, $cm^3/h/in.$ of nominal pipe size | 3 | 3 | 3 |

Table 1.3-1
NSSS VENDOR SUPPLIED COMPONENTS COMPARISON WITH SIMILAR FACILITY DESIGNS
(Sheet 29 of 33)

| Chapter | Valves | | KNU 5 & 6 | Maanshan | Shearon Harris |
|---|---|---|---|---|---|
| 6 | | Backseat leakage (when open), $cm^3/h/in.$ of stem diameter | 1 | 1 | 1 |
| | b. Gate valves | Disc leakage, $cm^3/h/in.$ of nominal pipe size | 3 | 3 | 1 |
| | | Backseat leakage (when open), $cm^3/h/in.$ of stem diameter | 1 | 1 | 1 |
| | c. Check valves | Disc leakage, $cm^3/h/in.$ of nominal pipe size | 3 | 3 | 3 |
| | d. Diaphragm valves | Disk leakage | None | None | None |
| | e. Pressure relief valves | Disk leakage, $cm^3/h/in.$ of nominal pipe size | 3 | 3 | 3 |
| | f. Accumulator | Disk leakage, $cm^3/h/in.$ of nominal pipe size | 3 | 3 | 3 |

Table 1.3-1

NSSS VENDOR SUPPLIED COMPONENTS COMPARISON WITH SIMILAR FACILITY DESIGNS
(Sheet 30 of 33)

| Chapter | Chapter Title System/Component | KNU 5 & 6 | Significant Similarities to MNPS | Significant Differences with MNPS |
|---|---|---|---|---|
| 7 | Instrumentation and controls | | | |
| | Reactor trip system | Section 7.2 | System functions are similar | None |
| | Engineered safety features systems | Section 7.3 | System functions are similar | None |
| | Systems required for safe shutdown | Section 7.4 | System functions are similar | None |
| | Safety-related display instrumentation | Section 7.5 | Parametric display is similar | None |
| | Other systems required for safety | Section 7.6 | Operational functions are similar | None |
| | Control systems not required for safety | Section 7.7 | Operational functions are similar | None |

Table 1.3-1

NSSS VENDOR SUPPLIED COMPONENTS COMPARISON WITH SIMILAR FACILITY DESIGNS
(Sheet 31 of 33)

| Chapter | System/Parameter | KNU 5 & 6 | Maanshan | Shearon Harris |
|---|---|---|---|---|
| 9 | CVCS design | Subsection 9.3.4 | | |
| | General | | | |
| | Seal water supply flow rate, for three reactor coolant pumps, nominal, gpm | 24 | 24 | 24 |
| | Seal water return flow rate, for three reactor coolant pumps, nominal, gpm | 9 | 9 | 9 |
| | Leakage flow: | | | |
| | Normal, gpm | 75 | 75 | 75 |
| | Maximum, gpm | 120 | 120 | 120 |
| | Charging flow (excludes sealwater): | | | |
| | Normal, gpm | 60 | 60 | 60 |
| | Maximum, gpm | 105 | 105 | 105 |
| | Temperature of letdown reactor coolant entering system, °F | 560 | 560 | <560 |

Table 1.3-1

NSSS VENDOR SUPPLIED COMPONENTS COMPARISON WITH SIMILAR FACILITY DESIGNS
(Sheet 32 of 33)

| Chapter | System/Parameter | KNU 5 & 6 | Maanshan | Shearon Harris |
|---|---|---|---|---|
| 9 | Temperature of charging flow directed to RCS, °F | $\geq$485 | $\geq$485 | >485 |
| | Temperature of effluent directed to boron recycle system, °F | 115 | 115 | 115 |
| | Centrifugal charging pump bypass flow (each), gpm | 60 | 60 | 60 |
| | Amount of 4% boric acid solution required to meet cold shutdown requirements shortly after full power operation, gal | 11,300 | 11,300 | 11,300 |

Table 1.3-1

NSSS VENDOR SUPPLIED COMPONENTS COMPARISON WITH SIMILAR FACILITY DESIGNS
(Sheet 33 of 33)

| Chapter | Chapter Title System/Component | References | Significant Similarities | Significant Differences |
|---|---|---|---|---|
| 11.0 | Radioactive waste management | | | |
| | Source terms | Section 11.1 | Maanshan and Shearon Harris | None |
| 14.0 | Initial tests and operation | Chapter 14 | Similar to Maanshan and Shearon Harris | None |
| 15.0 | Accident analyses | Chapter 15 | Similar to Maanshan and Shearon Harris | This section has been written to conform to USNRC R.G. 1.70 Rev. 02<br><br>This section has been written to conform to USNRC Standard Technical Specifications, Rev. 4 |

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Table 1.3-2

PLANT COMPONENTS OTHER THAN NSSS COMPARISON WITH SIMILAR FACILITY DESIGNS (Sheet 1 of 3)

| Chapter No. | System/Parameter | References | KNU 5 & 6 | Maanshan | SNUPPS | Pebble Springs |
|---|---|---|---|---|---|---|
| 3 | Containment type | Section 3.8 | Steel-lined post-tensioned pre-stressed concrete cylinder, hemispherical dome roof | Steel-lined post-tensioned pre-stressed concrete cylinder, hemispherical dome roof | Steel-lined post-tensioned pre-stressed concrete cylinder, hemispherical dome roof | Steel-lined post-tensioned pre-stressed concrete cylinder, hemispherical dome roof |
| | Leak rate, %/d | | 0.1 (24 h), 0.05 (after 24 h) | 0.1 (24 h), 0.05 (after 24 h) | 0.2 (24 h), 0.01 (after 24 h) | 0.1 |
| | Design pressure, psig | | 60 | 60 | 60 | 60 |
| | Free volume, $10^6$ $ft^3$ | | 2.15 | 2.0 | 2.50 | 2.45 |
| | Diameter/height, ft | | 130/195 | 130/195 | 140/205 | 130/225 |
| 6 | Containment spray | Paragraph 6.2.2.1 | | | | |
| | Number of pumps | | 2 | 2 | 2 | 2 |
| | Design capacity, ea gal/min | | 3,000 (Injection) 3,650 (Recirculation) | 3,000 (Injection) 3,650 (Recirculation) | 3,165 (Injection) 3,750 | 3,350 |
| | Spray additive | | NaOH | NaOH | NaOH | Hydrazine |
| | Containment coolers | Paragraph 6.2.2.2 | | | | |
| | Type | | Fan coolers | Fan coolers | Fan coolers | Fan coolers |
| | Number of units | | 4 | 4 | 4 | 6 |
| | Capacity, $10^6$ btu/h | | 50 | 50 | 110 | 97 |
| 8 | Onsite power systems, ac | Subsection 8.3.1 | | | | |
| | Generator prime mover | | Diesel engine | Diesel engine | Diesel engine | Diesel engine |
| | Number of units | | 2 | 2 | 2 | 2 |
| | Capacity, ea, kW | | 7,000 | 7,000 | 6,200 | 5,000 |
| 9 | Ultimate heat sink | Subsection 9.2.5 | | | | |
| | Type | | East Sea | Nan Wan Bay | (a) | Reservoir |
| | Backup | | None | None | | Spray pond |
| | Condensate Storage facility | Subsection 9.2.6 | | | | |
| | Capacity, $10^3$ gal | | 900 | 750 | 450 | 500 |
| | Plant fire protection | Subsection 9.5.1 | | | | |
| | Water source | | Tank storage | Tank storage | (b) | Reservoir |
| | Backup source | | None | None | | None |

Table 1.3-2

PLANT COMPONENTS OTHER THAN NSSS COMPARISON WITH SIMILAR FACILITY DESIGNS (Sheet 2 of 3)

| Chapter No. | System/Parameter | References | KNU 5 & 6 | Maanshan | SNUPPS | Pebble Springs |
|---|---|---|---|---|---|---|
| | Diesel generators | Subsection 9.5.4 | | | | |
| | Fuel oil storage capacity, per diesel operating at full power, days | | 7 | 7 | 7 | 7 |
| 10 | Turbine-generator | Section 10.2 | | | | |
| | output, MWe | | 993 | 951.8 | 1,186 | 1,313 |
| | Main steam supply | Section 10.3 | | | | |
| | Total steam flow, $10^6$ lb/h | | 12.29 | 12.29 | 15.13 | 15.30 |
| | Steam generator outlet pressure, psia | | 964 | 964 | 1,000 | 1,060 |
| | Steam generator outlet temp, °F | | 540.2 | 540 | 544.6 | 605 |
| | Main condensers | Subsection 10.4.1 | | | | |
| | Type | | Single pressure | Single pressure | Multiple pressure | Single pressure |
| | Pressure, in. HgA | | 1.5 | 2.54 | 2.06/2.56/3.22 | 1.5 |
| | Turbine bypass | Subsection 10.4.4 | | | | |
| | Capacity, % of rated load main steam flow | | 40 (to condenser)<br>30 (to atmosphere) | 32 (to condenser)<br>53 (to atmosphere) | 40 | 40 |
| | Circulating water | Subsection 10.4.5 | | | | |
| | Type | | Once-through, East Sea | Once-through, Nan Wan Bay | (c) | Closed system, reservoir |
| | Auxiliary feedwater | Subsection 10.4.9 | | | | |
| | pump prime movers | | 1 steam turbine, 2 electric motors | 1 steam turbine, 2 electric motors | 1 steam turbine, 2 electric motors | 2 diesel engines, 2 electric motors |
| | Rated flowrate, ea gal/min | | 1082 (turbine), 554 (motor) | 1084 (turbine), 554 (motor) | 1145 (turbine), 575 (motor) | 600 (engine), 600 (motor) |

Table 1.3-2

PLANT COMPONENTS OTHER THAN NSSS COMPARISON WITH SIMILAR FACILITY DESIGNS (Sheet 3 of 3)

| Chapter No. | System/Parameter | References | KNU 5 & 6 | Maanshan | SNUPPS | Pebble Springs |
|---|---|---|---|---|---|---|
| 11 | Radioactive waste management | | | | | |
| | Liquid waste systems tankage | Section 11.2 | 95,000 gal | 95,000 gal | 65,000 gal | 223,000 gal |
| | Gaseous waste systems holdup time(MIN.) | Section 11.3 | 45d | 45d | NA (zero release system) | 45d |
| | system type | | Charcoal delay | Charcoal delay | Decay tanks | Decay tanks |
| | Solid waste management solidification agent | Section 11.4 | Cement | Cement | Cement | Undecided |

a. Ultimate heat sink (SNUPPS)

Callaway (2 unit site) - Two 4-cell mechanical draft cooling towers with a retention pond as makeup source
Sterling - Once-through system utilizing Lake Ontario
Wolf Creek - Cooling pond

b. Plant fire protection (SNUPPS)

Callaway - Two 300,000 gallon storage tanks; water from supply to potable water
Sterling - Directly from Lake Ontario by fire pumps and pressurizing pump at circulating water system intake
Wolf Creek - Circulating water screen house intake bay

c. Circulating water (SNUPPS)

Callaway - Closed system, natural draft cooling tower, flowrate 550 x $10^3$ gal/min
Sterling - Once-through system, Lake Ontario, flowrate 795 x $10^3$ gal/min
Wolf Creek - Closed system, man-made lake, flowrate 550 x $10^3$ gal/min

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Table 1.3-3

SIGNIFICANT CHANGES FROM THE PSAR
(Sheet 1 of 3)

| FSAR Change | FSAR Chapter/ Section | Reason for Change |
|---|---|---|
| 1. Fuel design | 4.0, 15.0 & 16.0 | Shorter burnable poison rods and reduced number of burnable poison rods |
| 2. Safety grade cold shutdown added | 5.4, 6.3, 9.3 | Regulatory Guide 1.139 adopted |
| 3. Cold over-pressure mitigation | 5.2, 7.6, 16 | Provides additional RCS overpressure protection during low temperature water solid operation |
| 4. Fuel loading design change | 4 | To reduce core radial leakage |
| 5. Automated sump suction valve opening | 6.3 | Post-accident switch-over from ECCS injection to recirc-ulation improved. |
| 6. Diesel gener-ator output increased from 6,000 to 7,000 kW | 8.3 | Increased power requirements |
| 7. Fuel storage capacity | 9.1 | Increase capacity |
| 8. Condensate water storage decreased from 1,500,000 to 900,000 gallons | 9.2.6 | Incorporation of safety grade cold shutdown |
| 9. Addition of a firewater system designed to withstand SSE loads | 9.5.1 | Assures availability of water for fighting fires following an SSE |

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Table 1.3-3

SIGNIFICANT CHANGES FROM THE PSAR
(Sheet 2 of 3)

| FSAR Change | FSAR Chapter/ Section | Reason for Change |
|---|---|---|
| 10. Additional of a startup feedwater pump | 10.4 | To minimize use of auxiliary feedwater system during normal plant startup and shutdown. |
| 11. Seismic Category I instrument air storage tanks added | 9.3 | Implement Reg. Guide 1.139 |
| 12. Second steam generator PORV added per main steam lines | 10.3 | Incorporation of safety grade cold shutdown |
| 13. LRS second evaporator added | 11.2 | To optimize system availability and performance |
| 14. Fuel leak detection system | 9 | Detection of clad leakage |
| 15. Reactor coolant purity control system | 9 | Better chemistry control of reactor makeup water and recycled boric acid |
| 16. Metal impact monitoring system | 4 | Detects objects or metal in the RCS |
| 17. Revised control rod pattern | 4 | To insure nonpositive moderator coefficient 433 |
| 18. Full load rejection capability | 5, 10 | Provides capability for turbine trip from full load without reactor trip |

Table 1.3-3

SIGNIFICANT CHANGES FROM THE PSAR
(Sheet 3 of 3)

| FSAR Change | FSAR Chapter/ Section | Reason for Change |
|---|---|---|
| 19. Critical leak monitoring system | 5 | Detects vibration in RCS piping & components |
| 20. Automatic steam generator level control | 7, 10 | Automatic control of steam generator water level at low power |
| 21. Auxiliary shut-down panel design changed to two redundant panels and upgraded to meet Seismic Category 1 critiera | 7 | To meet separation single failure, and SSE criteria. |
| 22. Control isolation devices added to train is auxiliary shutdown panel | 7 | Protect one train of of hot shutdown equipment |
| 23. RWST water storage increased from 450,000 to 540,000 gallons and additional level instrumentation installed. | 6, 9 | Increase capacity to ensure adequate water for ECCS operation |
| 24. High density concrete added to auxiliary building shielding. | 12 | Improve effectiveness of shielding. |

## 1.4 IDENTIFICATION OF AGENTS AND CONTRACTORS

### 1.4.1 APPLICANT - KOREA HYDRO & NUCLEAR POWER COMPANY

KOREA HYDRO & NUCLEAR POWER COMPANY(KHNP), the sole government organized power utility in Korea, has total responsibility for construction and operation of KRN 3&4. KHNP has a long history of building and operating conventional electric generating plants. The overall KHNP organization is shown in figure 13.1-1. The home office of KHNP is in Gyeongju, Gyeongsangbuk-Do which is located about 33 miles north- 554
northeast of the Kori Nuclear Power site. KHNP is the sole applicant for the construction permit and facility license for KRN 3 & 4 .

KHNP, as owner, is responsible for the design, construction, and operation of these units. Overseas Bechtel, Incorporated, (Bechtel) assisted KHNP in engineering, procurement, and construction management. Westinghouse Electric Corporation" supplied the nuclear steam supply systems and provided the 390
services for nuclear fuel fabrication. GEC Turbine Generators, Ltd., supplied the turbine and Hitachi supplied the generators.

The Nuclear Power Construction Department (NPCD) of KHNP is responsible for the construction of KRN 3 & 4, and the Nuclear Power Generation Department (NPGD) of KHNP is responsible for the safe operation and maintenance of the units in compliance with technical specifications and other applicable requirements. The organization of NPCD and NPGD is shown in figure 13.1-1. Plant organization and responsibilities for plant operation are described in subsections 13.1.2 and 13.1.3. Plant organization and responsibilities during preoperational testing, startup, and initial operation are described in chapter 14.

### 1.4.2 BECHTEL'S QUALIFICATIONS AND EXPERIENCE

Overseas Bechtel, Incorporated, has been retained by KHNP to provide architectural and engineering services, including 390
procurement, construction, and technical direction and services for KRN 3 & 4.

Bechtel has been continuously engaged in construction and engineering activities since 1898. Since the close of World War II, Bechtel has placed strong emphasis on electrical power-generation projects. During this period, Bechtel has been responsible for the design and/or construction of over 237 thermal generating units, representing more than 126,000,000 kW of new generating capacity. Of this number, a nuclear capacity of more than 65,000,000 kW has been, or is being, engineered.

The ratings of thermal generating plants designed by Bechtel range to 1,470,000 kW per unit and include most types of station designs and arrangements, such as reheat and non-reheat, indoor and outdoor stations, single and multiple units, and wide ranges of steam conditions up to 3500 psig, 1050/1000F. Also, some of the larger units are fully automated and computer controlled. The majority of contracts for these facilities provided Bechtel with complete responsibility for engineering, construction, and startup.

For more than 26 years Bechtel has been actively working on nuclear projects involving power plants, as well as such facilities as nuclear accelerators, research laboratories, hot cells, experimental reactors, and nuclear fuel processing plants. The responsibilities have covered design, construction, startup, site surveys, license applications, feasibility studies, and equipment procurement.

### 1.4.3 WESTINGHOUSE'S QUALIFICATIONS AND EXPERIENCE AS A SUPPLIER OF NUCLEAR STEAM SUPPLY SYSTEMS

Westinghouse Electric Corporation (Westinghouse) is responsible for supplying the NSSS and fuel for Korea Units 5 & 6.

Westinghouse has designed, developed, and manufactured nuclear power facilities since the 1950s, beginning with the world's first large central station nuclear power plant (Shippingport), which has produced power since 1957. Completed or presently contracted commercial nuclear capacity totals in excess of 97,000 megawatts. Westinghouse pioneered new nuclear design concepts, such as chemical shim control of reactivity and the rod cluster control concept, throughout the last two decades. Westinghouse manufacturing facilities include the largest commercial nuclear fuel fabrication facility in the world, and the world's most modern heat transfer equipment production facility as well as other facilities producing NSSS components. Table 1.4-1 lists all Westinghouse pressurized water reactor (PWR) plants to date, including those plants currently under construction or on order.

The U.S. Nuclear Regulatory Commission (NRC) and the Edison Power Research Institute have contracted with Westinghouse for research into NSSS-related activities. Westinghouse experience was also utilized by the NRC and Metropolitan Edison immediately following the Three Mile Island Unit 2 accident and remains as a heavy participant with the Westinghouse Owner's Group of utilities in addressing the NRC's action plan for corrective actions.

### 1.4.3.1 Plants in Operation

Westinghouse PWR plants in operation are as follows:

A. Shippingport

Shippingport was the world's first large central station nuclear power plant. The reactor plant was designed by the Bettis Atomic Power Laboratory, which is operated by Westinghouse under an NRC contract. Shippingport's PWR has produced power for Duquesne Light Company since December 1957.

B. Yankee-Rowe

Singled out by the NRC as a "Nuclear Success Story," Yankee-Rowe went on line in November 1960. Owned and operated by the Yankee Atomic Electric Company, Yankee-Rowe has progressed from an initial rating of 120 MWe to its present 175 MWe rating. Westinghouse supplied the NSSS and the turbine generator.

C. Trino Vercellese (Enrico Fermi)

The Trino Vercellese nuclear plant was one of the first Westinghouse-designed plants to incorporate chemical shim control of reactivity. Chemical shim has since become a standard feature of Westinghouse PWR control. Trino Vercellese achieved initial criticality in June 1964 and began power operation in October 1964. The plant is rated at 260 MWe.

D. Chooz (Ardennes)

The Chooz plant is unique in that the Westinghouse PWR and its auxiliaries are housed in man-made caverns. Ardennes, a joint Franco-Belgian undertaking, owned and operated by the Societa d'Energie Nucleaire Franco-Belge des Ardennes (SENA), is located in France near the French-Belgian border. Chooz achieved initial criticality in October 1966 and began power operation in 1967.

E. San Onofre No. 1

San Onofre No. 1 employs the Westinghouse-developed rod cluster control concept which has since become a standard feature of the Westinghouse PWR. Owned by the Southern California Edison Company and the San Diego Gas and Electric Company, the 450 MWe plant is located near San Clemente, California. Westinghouse supplied the NSSS and the turbine generator. Initial criticality was achieved in June 1967, and power operation began in January 1968.

F. Haddam Neck (Connecticut Yankee)

Owned and operated by the Connecticut Yankee Atomic Power Company, this plant went critical in July 1967 and attained full power operation in December 1967. Like San Onofre No. 1, the plant employs rod cluster control in conjunction with chemical shim control. Westinghouse supplied the NSSS and the turbine generator. The plant has been uprated to 575 MWe.

G. Jose Cabrera - (Zorita)

The Jose Cabrera station is located near Zorita, Spain. The 153 MWe plant employs rod cluster control, chemical shim control and a Zircaloy-clad core. Construction began in mid-1965, and power operation began in 1968. Jose Cabrera is owned and operated by the Union Electrica, S.A., a Spanish utility.

H. Beznau No. 1 and No. 2

Beznau No. 1, Switzerland's first commercial nuclear power plant, achieved initial criticality in June 1969 and supplied power to the system in July 1969. The 350 MWe plant was designed and constructed by the Westinghouse-Brown Boveri Consortium for the owner/operator utility, Nordostschweizerische Kraftwerke A.G. The plant started producing power less than 4 years after award of the plant contract. Beznau No. 2 achieved criticality in October 1971 and began commercial operation in early 1972.

I. Robert Emmett Ginna

The Robert Emmett Ginna Plant, owned and operated by Rochester Gas and Electric Corporation, is located in New York on the south shore of Lake Ontario. Westinghouse supplied the 490 MWe plant on a turnkey basis. Construction began in April 1966 with initial criticality being achieved in November 1969 (just 42 months after start of construction). Power was supplied to the system in December 1969.

J. Mihama No. 1 and Takahama No. 1

These plants are owned by the Kansai Electric Power Company, Inc. Mihama No. 1 is a two-loop, 320 MWe unit and marks the beginning of a line of Westinghouse PWRs supplying the generation needs of the Far East. Westinghouse International Company was the prime contractor for the Mihama project, supplying the NSSS engineering, nuclear fuel, and some major system components. Mihama No. 1 required only 44 months from the start of site construction to first power production in August 1970. Takahama No. 1 is a three-loop, 780 MWe unit. Initial criticality was achieved in March 1974.

K. H.B. Robinson No. 2

This plant is a three-loop, 707 MWe unit which was built on a turnkey basis for the Carolina Power and Light Company. The plant is located at a site near Hartsville, South Carolina, on a man-made cooling lake. The construction permit was granted in April 1967. The plant achieved criticality in August 1970 and first power to system in October 1970.

L. Point Beach No. 1 and No. 2

The Point Beach Project consists of two 497 MWe units, which were built on a turnkey basis for the Wisconsin Michigan Power Company and the Wisconsin Electric Power Company. The plants are located near Two Creeks, Wisconsin, 90 miles north of Milwaukee on Lake Michigan. This was the first two-unit station to utilize many common facilities and shared auxiliary systems. The construction permit for Point Beach No. 1 was granted in July 1967 with initial criticality and first power to the system in November 1970. Point Beach No. 2 went critical in May 1972 and was available for commercial operation in October 1972.

M. Surry No. 1 and No. 2

The Surry Power Station, two three-loop 822 MWe units, is owned by the Virginia Electric and Power Company. The James River Station is about 30 miles from Norfolk, Virginia. First criticality on Surry No. 1 was achieved in July 1972. Commercial operation began in September 1972. Initial criticality on Surry No. 2 was achieved in March 1973.

N. Turkey Point No. 3 and No. 4

Florida Power and Light Company is the owner of a four-unit station on Biscayne Bay, Florida. Turkey Point Nos. 3 and 4 of the station are three-loop, 745 MWe plants. Commercial status for Turkey Point No. 3 was achieved in December 1972. Initial criticality for Turkey Point No. 4 was achieved in June 1973.

O. Indian Point No. 2 and No. 3

Consolidated Edison Company of New York operates three nuclear units located in Buchanan, New York; two (Units 1 and 2) are owned by the company and one (Unit No. 3) is owned by the Power Authority of the state of New York. Units 2 and 3 are Westinghouse PWRs rated at 873 and 965 MWe respectively. Indian Point No. 2 achieved initial criticality in May 1973 and Indian Point No. 3 achieved initial criticality in April 1976.

P. Prairie Island No. 1 and No. 2

Northern States Power Company is the owner of these two-loop, 530 MWe units located in Welch, Minnesota. Initial criticality was achieved in December 1973 for Prairie Island No. 1, and in December 1974 for Prairie Island No. 2.

Q. Zion No. 1 and No. 2

Commonwealth Edison Company is the owner of these two four-loop, 1050 MWe units. The units are located on Lake Michigan near Zion, Illinois. Initial criticality was achieved in June 1973 for Zion No. 1 and in December 1973 for Zion No. 2.

R. Kewaunee

Wisconsin Public Service Corporation, Wisconsin Power and Light Company, and Madison Gas and Electric Company are the owners of this two-loop, 541 MWe plant located in Kewaunee, Wisconsin. Initial criticality was achieved in March 1974.

S. Ringhals No. 2

Statens Vattenfallsverk (SSPB) is the owner of this three-loop, 822 MWe unit located in Sweden. Initial criticality was achieved in June 1974.

T. Donald C. Cook No. 1 and No. 2

Indiana and Michigan Electric Company is the owner of this four-loop, 1090 MWe plant located in Bridgman, Michigan. This plant is the first to use the Westinghouse Ice Condenser Containment design. Initial criticality was achieved in January 1975 for Unit 1 and March 1978 for Unit 2.

U. Trojan

This four-loop, 1130 MWe plant is jointly owned by Portland General Electric Company, Eugene Water and Electric Board, and Pacific Power and Light Company. In addition to being the first commercial nuclear plant to operate in the Pacific Northwest (located on the Oregon shore of the Columbia River near Rainier, Oregon), Trojan is the first 17 x 17 fuel-rod-per-assembly plant to achieve criticality. Initial criticality was achieved in December 1975.

V. Beaver Valley No. 1

This three-loop, 852 MWe plant is jointly owned by Duquesne Light Company, Ohio Edison Company, and Pennsylvania Power Company. Beaver Valley No. 1 is located on the Ohio River, 22 miles northwest of Pittsburgh, Pennsylvania. Commercial operation began in early 1976.

W. Salem No. 1 and No. 2

Salem No. 1 and 2, owned jointly by the Public Service Electric and Gas Company, Philadelphia Electric Company, Atlantic Electric Company, and Delmarva Power and Light Company, is located on Artificial Island, a man-made peninsula in Salem County, New Jersey. The 1090 MWe, four-loop plant achieved initial criticality for Unit 1 in late 1976 and Unit 2 achieved criticality in August 1980.

X. North Anna No. 1 and No. 2

Virginia Electric and Power Company owns the two approximately 907 MWe (net) plants located 40 miles north of Richmond, Virginia on Lake Anna. Unit 1 achieved criticality in May 1978 and Unit 2 achieved criticality in June 1980.

Y. Joseph M. Farley No. 1 and No. 2

The two 899 MWe (net) Alabama Power Company units are located at Dothan, Alabama, (approximately 180 miles south-southwest of Atlanta, Georgia. Unit 1 achieved criticality in August 1977, Unit 2 achieved criticality in February 1981.

Z. Sequoyah No. 1 and No. 2

The two units approximately 1148 MWe (net) are located on the Tennessee River near Chattanooga, Tennessee. These units are owned by Tennessee Valley Authority. Sequoyah No. 1 received a full power license in September 1980.

### 1.4.3.2 Westinghouse Facilities

Westinghouse, in its effort to plan for the future, has developed a broad range of facilities to satisfy the needs of the nuclear industry. The following paragraphs briefly describe these facilities:

A. Columbia Plant, Nuclear Fuel Division

The Columbia Plant is capable of performing all operations necessary to manufacture finished nuclear fuel assemblies. These operations include conversion of uranium hexafluoride to uranium dioxide powder, fabrication of fuel assembly grids, complete pellet loading, and final fabrication of assemblies. The plant, located at Columbia, South Carolina, began full production in early 1970. The Columbia Plant is the largest commercial nuclear fuel fabrication facility in the world.

B. Tampa Division*

The Tampa Division Plant is the world's most modern heat transfer equipment production facility. The plant has 236,000 square feet of working space with two manufacturing aisles for the production of steam generators and pressurizers. Transportation facilities include four railroad spurs and a complete barge slip and dock facility for water shipment to all parts of the world. The Tampa Division Plant made its first steam generator and pressurizer shipment in September 1969.

*All manufacturing from the Tampa Division Plant has been transferred to the Westinghouse Pensacola Division.

C. Pensacola Division

The Pensacola Division Plant, located on Escambia Bay on the northwest coast of Florida, is a new 140,000-square-foot manufacturing plant for producing precision reactor vessel internals. Contributing to the precision manufacturing capability is an environmental control system which minimizes year-round temperature changes throughout the shop area. Transportation facilities of the plant include a railroad spur which permits loading and unloading inside the shop, and access to barge loading facilities on Escambia Bay. Pensacola shipped its first package of reactor internals in July 1970.

D. Cheswick Plant, Electro-Mechanical Division

The Electro-Mechanical Division was established in Cheswick, Pennsylvania, in 1953 to manufacture canned motor primary coolant pumps for nuclear reactors. Today, the product line has expanded to include shaft seal pumps (reactor coolant pumps), valves from 4 inches to 31 inches, and control rod drive mechanisms, essential components of the Westinghouse PWR. The facility occupies 250,000 square feet and now contains the most modern facilities available for the production and testing of nuclear plant components.

E. Specialty Metals Division

The Specialty Metals Division located in Blairsville, Pennsylvania, was completed in late 1967. Several essential PWR component processes are accomplished at Blairsville, including: the precision manufacture of Inconel tubing for steam generators, and the complete processing of zircaloy seamless tubing for nuclear fuel cladding. At Blairsville, complete quality control facilities are utilized for the evaluation and analysis of all specialty metal products used in Westinghouse nuclear systems.

F. Westinghouse Nuclear Center

The headquarters of Westinghouse Nuclear Energy Systems is located just east of Pittsburgh, in Monroeville, Pennsylvania. Operating primarily as a headquarters and engineering facility, the complex houses many of the divisions which encompass Westinghouse's nuclear activities associated with the electric utility industry.

G. Zion Nuclear Training Center

The Westinghouse Electric Corporation and the Commonwealth Edison Company of Chicago have built and are operating a nuclear training center at Zion, Illinois. The 28,000-square-foot training center contains classrooms, a training reactor, training material center, video recording facilities, and multi-plant nuclear power plant simulators. Westinghouse staffs and operates the center, supplies all the equipment required, and is responsible for the development and presentation of all training programs. Commonwealth Edison provided the building, access to the Zion nuclear units for conducting in-plant observation training, and advises and assists Westinghouse in developing training programs.

### 1.4.4 GENERAL ELECTRIC COMPANY'S QUALIFICATIONS AS A SUPPLIER OF TURBINES

The General Electric Company (GEC), Limited, designed, fabricated, and delivered the turbine , as well as provided technical assistance for the installation and startup of this equipment. 390

GEC Limited has a long history in the application of turbine generators in nuclear power plants.

#### 1.4.4.1 Contractor Description

GEC Power Engineering Limited, one of five functional management subsidiaries, is the largest specialist group of its kind in Europe and provides for every facet of power generation, transmission, and distribution. The Group employs about 25,000 people and its annual sales approach $500 million. The GEC Power Engineering Group provides a complete capability in the power field.

The Group, both directly and through AEI and English Electric, has had extensive experience in the design and construction of complete nuclear power plants. It was the lead contractor in the Consortia which built Tokai Mura, Hunterston A, Hinckley Point A, Sizewell A, and Wylfa nuclear power stations. Additionally, the Group has many year's experience as lead contractor for the design and construction of complete fossil, hydro, and gas turbine stations in all continents of the world.

GEC generating plant and ancillary equipment is installed in fossil and nuclear fueled power stations throughout the world. GEC is the major supplier of 660 MW turbine generators to Britain's home electricity generating boards, and, in the overseas market, the 1200 MW sets for the U.S.A. are among the largest produced anywhere.

The Group, through its trading subsidiaries, is actively engaged in a number of power station contracts in the U.S.A., Canada, Africa, the Middle East, Scandinavia, and the Far East.

GEC, an international leader in power transmission and distribution, offers a wide range of circuit breakers and switchgear: also transformers and reactors, up to the largest capacities required. In addition, GEC supplies completely engineered schemes for dc transmission, and the compensation of ac transmission and distribution networks using either synchronous or static compensation and employing GEC's unique designs of saturated reactor.

GEC Turbine Generators Limited is the principal member of the GEC Power Engineering Group. The GEC Turbine Generator Company is the major supplier to the United Kingdom Central Electricity Generating Board, itself the world's largest interconnected electricity supply system. It has a long history of technical development, innovation, and achievement in large-sized units and has 57 units of 500 MW or greater unit size installed or on order, totaling 34,334 MW.

The company has a similarly impressive record in the nuclear field and has 43 units totaling 15,094 MW installed or on order for nuclear power stations in Britain, the U.S.A., Canada, Sweden, Japan, and India. Since the company's first nuclear turbine generator started operation 15 years ago, 289 unit years of nuclear turbine operating experience have been accumulated in association with a variety of reactor types, Magnox, Advanced Gas Cooled, Pressurized water, Boiling Water, Candu, Steam Generating Heavy Water, High Temperature Gas Cooled and Fast Reactors. Nuclear units of 660 MW are already operational and 12 more units of 600 to 1200 MW are in advanced stages of constructions.

The turbine being supplied by GEC for this project is one of GEC's current family of 600 MW to 1200 MW, 1800 rpm turbine generator designs for nuclear power stations. Existing contracts for this class of equipment are the 600 MW Ko-Ri I, 650 MW Ko-Ri II and the 1200 MW Enrico Fermi and San Onofre machines. 390

The 1200 MW unit for Enrico Fermi (Detroit, U.S.A.) was shipped in October 1974, and the turbine for the first San Onofre station was shipped in March 1977. The 600 MW unit for Ko-Ri I was shipped in 1975 and is currently in use.

The extensive experience of GEC in the design, supply, and construction of nuclear turbine generators and associated mechanical and electrical auxiliaries, and in the management of complete nuclear and thermal power stations, coupled with the accumulated experience of power station construction in Korea, makes GEC well qualified to participate in this nuclear project.

1.4.4.2 Contractor Experience

1.4.4.2.1 General Design Parameters

In the context of a reference plant already manufactured or in service, it is pertinent to assess various design criteria in order to demonstrate that component design for the KRN 3 & 4 machines falls within existing experience. Such design criteria are listed below for the following principal components: 390

A. High pressure turbine

B. Low pressure turbine

C. Control interface

D. Moisture separators and reheaters

The criteria listed in tables 1.4-2 and 1.4-3 cover details such as sizes, weights, flow parameters, and stress parameters as appropriate.

1.4.4.2.1.1 High Pressure Turbine. Three parameters may be considered in assessing the relevance of one design relative to another:

A. Volumetric flow at inlet to the blading, measured as $\frac{GV}{N}$, where:

G = mass flow at turbine inlet

V = specific volume at inlet

N = number of flows in HP cylinder

This parameter has relevance to the blading area required, that is, to blade heights and mean diameters.

B. Average torque developed in each state (Ti), which has relevance to the blading strength.

C. A "sealing duty coefficient" S, or pressure force per unit of casing length, measured as Di x$\Delta$p,

where:

Di = casing internal diameter

$\Delta$p = pressure drop across casing

both in the region of the first stage

This has relevance to the sealing duty of the horizontal joint.

These parameters for the reference installations (Kori 1, Ringhals 1 and 2, and San Onofre 2 and 3) are listed in table 1.4-4.

Tables 1.4-5 and 1.4-6 show comparisons of principal rotor and blading dimensions for the reference installations plus the IP rotor for Peterhead 660 MW, 3000 rpm fossil-fired machine, which is representative of 11 similar machines.

1.4.4.2.1.2 Low Pressure Turbine. The low pressure turbines for KNU 5 & 6 are closely similar to those in service at Kori 1 and under construction for Kori 2, Enrico Fermi 2, and San Onofre 2 and 3. In addition, the reheat pressures and temperatures associated with these water-cooled reactor plants have been selected to ensure that the low pressure cylinder exhaust wetness is comparable with the wetness in conventional fossil-fired plant. The vast accumulation of experience on material selection and protective measures obtained from all of these plants has been applied to the turbines for KNU 5 & 6.

1.4.4.2.2 Relevant Design Rating Parameters

As the low pressure turbines offered are very similar or even virtually identical geometrically and in the choice of materials to the low pressure turbines in two of the reference installations, many relevant design rating parameters concerning centrifugal stressing, rotor dynamics, and blading vibration need not be considered here.

The remaining relevant design rating parameters which are variable with the steam cycle and design applications are:

G - steam flow through last stage of a low pressure turbine

or $\left(\frac{G}{Ae}\right)$ - last blade loading

where Ae - annulus area of last stage blading in the low pressure turbine (2 flows)

With regard to the likelihood of blading tip erosion, GEC has developed for design assessment purposes a sophisticated "erosion criterion number." This takes into account a large number of variables which play a part in the very complex mechanism of droplet formation and their deposition on stationary blading, subsequent stripping off and breakup of large drops followed by impact against the rotating blading. However, since the reference installations employ virtually identical low pressure turbines, it is possible in our relative comparisons to reduce the complexities of the "erosion criterion number" to a very simple expression.

Relative erosion criterion: $\propto G \cdot \frac{x}{p^{\alpha} \cdot c^{\beta}}$

where

G - last stage steam flow - (as before in this section)

x - wetness at inlet to the last stage, percent

p - pressure upstream of the last moving blade (at tip)

c - velocity of steam discharging from last stage stationary blading

$\alpha$ - at least 2.0, but probably between 2.0 and 3.0

<u>Kori 1</u> ) virtually identical complete
<u>San Onofre 2 and 3</u> ) low pressure turbine

<u>Ringhals 1 and 2</u> 3000 rpm turbine with comparable BWR cycle

The relevant comparative design rating parameters are summarized in tables 1.4-7, 1.4-8, and 1.4-9.

1.4.4.2.3 Separators and Reheaters

GEC has operating experience with moisture separator/reheaters of similar design on the Ringhals 400 MW BWR machines. The Ringhals units employ live steam heating only and have extended surface stainless steel tubes welded to stainless-steel-clad forged-steel tubeplates. Apart from some initial baffle damage the separator/reheaters on both machines have given satisfactory performance since 1975.

GEC has manufactured separator/reheaters in 2 x 50 percent units for San Onofre and Enrico Fermi in the U.S.A. to operate with PWR and BWR cycles, respectively. Both San Onofre and Enrico Fermi utilize mild steel extended surface tubing welded to forged steel tubeplates.

1.4.4.2.4 Relevant Design Rating Parameters

Generally these are:

G - mass flow of wet steam entering the separator

x - inlet wetness of steam entering the separator

p - pressure of wet steam entering the separator

T - temperature of reheated steam leaving the reheater

$\Delta$T - approach temperature of reheated steam to live steam saturation temperature

A - total heat exchange surface of reheater

It is, of course, well known that separation efficiency increases as the pressure decreases.

One of the possible reference installations in which moisture separating and reheating equipment of GED's own design and manufacture is employed is San Onofre Units 2 and 3 and comparative design rating parameters are as shown in tables 1.4-10 and 1.4-11. 390

### 1.4.5 Hitachi's Qualification As a Supplier of Generators

Hitachi, Limited designed, fabricated, and delivered the turbine generator, as well as provided technical assistance for the installation and startup of this equipment.

Hitachi, Limited has a long history in the application of turbine generators in nuclear power plants.

#### 1.4.5.1 Contractor Description

Hitachi, Limited is the largest electrical machinery manufacturing company in Japan and provides for every facet of power generation, transmission, and distribution. The Group employs about 390,000 people and its consolidated annual sales approach 11,227,000 million JPY. Hitachi provides a complete capability in the power field including boiling water type nuclear power plants. 390

The Group has had extensive experience in the design and construction of complete nuclear power plants. It was the one of consortium contractor of Kashiwazaki - Kariwa unit 6 and 7 of Japanese first ABWRs, Hamaoka unit 5 and was the complete supplier of Shika Unit 2 nuclear power stations of 1350MWe ABWRs. Additionally, the Group has many years' experience as lead contractor for the design and construction of complete fossil, hydro, and gas turbine stations in all over of the world.

Hitachi generating plant and ancillary equipment is installed in fossil and nuclear-fueled power stations throughout the world. Hitachi is the major supplier of 1100 MWe and 1350MWe turbine generators in Japan and in the oversea market.

The Group, through its trading subsidiaries, is actively engaged in a number of power station contracts in the U.S.A., Canada, China, the Middle East, and Korea.

Hitachi, an international leader in power transmission and distribution, offers a wide range of circuit breakers and switchgear: also transformers and reactors, up to the largest capacities required. In addition, Hitachi supplies completely engineered schemes for dc transmission, and the compensation of ac transmission and distribution networks using either synchronous or Static Var Compensation system.

Hitachi has an impressive record in the nuclear field and has 20 units totaling about 18,200 MW installed or on order for nuclear power stations in Japan, Korea, Pakistan and China. Since the company's first nuclear turbine generator started operation 47
years ago, 347 unit years of nuclear turbine generator operating experience have been accumulated in association with a variety of reactor types, Pressurized Water, Boiling Water, Candu reactors. 4 Nuclear units of 1350 MW have been operated and 2 more units of 1350MW are in the stages of constructions in Japan.

The turbine generators being supplied by Hitachi for this project are Hitachi's current family of 700 MW to 1350 MW, 1800 rpm turbine generator designs for nuclear power stations. 390

The extensive experience of Hitachi in the design, supply, and construction of nuclear turbine generators and associated mechanical and electrical auxiliaries, and in the management of complete nuclear and thermal power stations, coupled with the accumulated experience of power station construction of the consortium partner of Doosan Heavy Industrial and Construction Ltd. in Korea, makes Hitachi well qualified to participate in this nuclear project

1.4.5.2 Contractor Experience

1.4.5.2.1 Generator

The generator of KRN 2 is of the same design except its rates capacity and cooler type of that of KRN 1 which has been tested at Hitachi Works and is operated successfully. In addition Hitachi has many experiences of the nuclear generators and the design of KRN 1 & 2 Generators is based on that of Hamaoka unit 3 & 4 generators in Japan.

The design of KRN 3 & 4 generators are also based on Hamaoka unit 3 & 4 generators, however the rated capacity, dimensions are different from those of KRN 1& 2, so the first one, KRN 4 Generator will be tested in Hitachi Works to confirm its required characteristics as same as KRN 1 generator. No development work of design parameters is required because Hitachi has been supplied many generators to nuclear power plants as shown in the below table and Hitachi design has been well proved of enough reliability.

Table 1.4-12 compares the principal design parameters for KRN 1, KRN 3 & 4, and Enrico Fermi.

Table: Hitachi supply list to nuclear power plants

| Country | Power Station | Capacity (MVA) | Speed (min-1) | Voltage (KV) | Service YeAR |
|---|---|---|---|---|---|
| Japan | Hamaoka-5 | 1570 | 1800 | 22 | 2005 |
| Japan | Shika-2 | 1540 | 1800 | 24 | 2006 |
| Japan | Shimane-3 | 1530 | 1800 | 22 | 2011 |
| Japan | Kashiwazaki Kariwa-6 | 1540 | 1500 | 27 | 1996 |
| Japan | Fukushima-11-2 | 1300 | 1500 | 19 | 1984 |
| Japan | Fukushima-11-4 | 1300 | 1500 | 19 | 1987 |
| Japan | Kashiwazaki Kariwa-5 | 1300 | 1500 | 19 | 1990 |
| Japan | Kashiwazaki Kariwa-4 | 1300 | 1500 | 19 | 1994 |
| Japan | Hamaoka-4 | 1280 | 1800 | 22 | 1993 |
| Japan | Hamaoka-3 | 1280 | 1800 | 22 | 1987 |
| Japan | Hamaoka-2 | 940 | 1800 | 18 | 1978 |
| Japan | Onagawa-3 | 920 | 1500 | 20 | 2002 |
| Japan | Fukushima- 1 -4 | 911 | 1500 | 17 | 1978 |
| Japan | Shimane-2 | 870 | 1800 | 15.5 | 1989 |
| China | Qinshan-111-1 | 817 | 1500 | 22 | 2003 |
| China | Qinshan-111-2 | 817 | 1500 | 22 | 2004 |
| Korea | Kori-1 | 749 | 1800 | 22 | 2005 |
| Japan | Hamaoka-1 | 626 | 1800 | 22 | 1976 |
| Japan | Shika-1 | 600 | 1800 | 20 | 1993 |
| Japan | Shimane-1 | 520 | 1800 | 18 | 1974 |
| Parkistan | Karachi | 163 | 3000 | 15 | 1969 |

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Table 1.4-1

WESTINGHOUSE PRESSURIZED WATER REACTOR NUCLEAR POWER PLANTS (Sheet 1 of 15)

| Plant | Owner Utility | Location | Scheduled Commercial Operation | MWe Net | Number of Loops |
|---|---|---|---|---|---|
| Shippingport | Duquesne Light Company | Pennsylvania | 1957 | 90 | 4 |
| Yankee-Rowe | Yankee Atomic Electric Company | Massachusetts | 1961 | 175 | 4 |
| Trino Vercellese (Enrico Fermi) | Ente Nazionale per l'Energia Elettrica (ENEL) | Italy | 1965 | 260 | 4 |
| Chooz (Ardennes) | Societe d'Energie Nucleaire Franco-Belge des Ardennes (SENA) | France | 1967 | 305 | 4 |
| San Onofre No. 1 | Southern California Edison Co.; San Diego Gas and Electric Co. | California | 1968 | 450 | 3 |
| Haddam Neck (Connecticut Yankee) | Connecticut Yankee Atomic Power Company | Connecticut | 1968 | 575 | 4 |
| Jose Cabrera-Zorita | Union Electrica, S. A. | Spain | 1969 | 153 | 1 |
| Beznau No. 1 | Nordostschweizerische Kraftwerke AG (NOK) | Switzerland | 1969 | 350 | 2 |

Table 1.4-1

WESTINGHOUSE PRESSURIZED WATER REACTOR NUCLEAR POWER PLANTS (Sheet 2 of 15)

| Plant | Owner Utility | Location | Scheduled Commercial Operation | MWe Net | Number of Loops |
|---|---|---|---|---|---|
| Robert Emmett Ginna | Rochester Gas and Electric Corporation | New York | 1970 | 490 | 2 |
| Mihama No. 1 | The Kansai Electric Power Company, Inc. | Japan | 1970 | 320 | 2 |
| Point Beach No. 1 | Wisconsin Electric Power Co.; Wisconsin Michigan Power Co. | Wisconsin | 1970 | 497 | 2 |
| H. B. Robinson No. 2 | Carolina Power and Light Co. | South Carolina | 1971 | 707 | 3 |
| Beznau No. 2 | Nordostschweizerische Kraftwerke AG (NOK) | Switzerland | 1972 | 350 | 2 |
| Point Beach No. 2 | Wisconsin Electric Power Co.; Wisconsin Michigan Power Co. | Wisconsin | 1972 | 497 | 2 |
| Surry No. 1 | Virginia Electric and Power Co. | Virginia | 1972 | 822 | 3 |
| Turkey Point No. 3 | Florida Power and Light Co. | Florida | 1972 | 745 | 3 |

Table 1.4-1

WESTINGHOUSE PRESSURIZED WATER REACTOR NUCLEAR POWER PLANTS (Sheet 3 of 15)

| Plant | Owner Utility | Location | Scheduled Commercial Operation | MWe Net | Number of Loops |
|---|---|---|---|---|---|
| Indian Point No. 2 | Consolidated Edison Company of New York, Inc. | New York | 1973 | 873 | 4 |
| Prairie Island No. 1 | Northern States Power Company | Minnesota | 1973 | 530 | 2 |
| Turkey Point No. 4 | Florida Power and Light Co. | Florida | 1973 | 745 | 3 |
| Surry No. 2 | Virginia Electric and Power Co. | Virginia | 1973 | 822 | 3 |
| Zion No. 1 | Commonwealth Edison Company | Illinois | 1973 | 1,050 | 4 |
| Kewaunee | Wisconsin Public Service Corp.; Wisconsin Power and Light Co.; Madison Gas and Electric Co. | Wisconsin | 1974 | 560 | 2 |
| Prairie Island No. 2 | Northern States Power Company | Minnesota | 1974 | 530 | 2 |
| Takahama No. 1 | The Kansai Electric Power Company, Inc. | Japan | 1974 | 781 | 3 |

Table 1.4-1

WESTINGHOUSE PRESSURIZED WATER REACTOR NUCLEAR POWER PLANTS (Sheet 4 of 15)

| Plant | Owner Utility | Location | Scheduled Commercial Operation | MWe Net | Number of Loops |
|---|---|---|---|---|---|
| Zion No. 2 | Commonwealth Edison Company | Illinois | 1974 | 1,050 | 4 |
| Beaver Valley No. 1 | Duquesne Light Company; Ohio Edison Company; Pennsylvania Power Company | Pennsylvania | 1976 | 852 | 3 |
| Doel No. 1 | Indivision Doel | Belgium | 1975 | 390 | 2 |
| Doel No. 2 | Indivision Doel | Belgium | 1975 | 390 | 2 |
| Donald C. Cook No. 1 | Indiana and Michigan Electric Company (AEP) | Michigan | 1975 | 1,090 | 4 |
| Donald C. Cook No. 2 | Indiana and Michigan Electric Company (AEP) | Michigan | 1978 | 1,090 | 4 |
| Indiana Point No. 3 | Power Authority of the State of New York (PASNY) | New York | 1976 | 965 | 4 |
| Kori No. 1 | Korea Hydro & Nuclear Power Co. | Korea | 1977 | 564 | 2 |
| Ringhals No. 2 | Statens Vattenfallsverk (SSPB) | Sweden | 1975 | 822 | 3 |

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Table 1.4-1

WESTINGHOUSE PRESSURIZED WATER REACTOR NUCLEAR POWER PLANTS (Sheet 5 of 15)

| Plant | Owner Utility | Location | Scheduled Commercial Operation | MWe Net | Number of Loops |
|---|---|---|---|---|---|
| Trojan | Portland General Electric Co.; Eugene Water and Electric Board; Pacific Power and Light Company | Oregon | 1976 | 1,130 | 4 |
| Almaraz No. 1 | Union Electrica, S. A.; Compania Sevillana de Electricidad, S. A.; Hidroelectrica Espanola, S. A. | Spain | 1981 | 902 | 3 |
| Diablo Canyon No. 1 | Pacific Gas and Electric Co. | California | 1981 | 1,084 | 4 |
| Joseph M. Farley No. 1 | Alabama Power Company | Alabama | 1977 | 829 | 3 |
| Lemoniz No. 1 | Iberduero, S. A. | Spain | 1984 | 902 | 3 |
| Salem No. 1 | Pacific Service Electric and Gas Company; Philadelphia Electric Co.; Atlantic Electric Co.; Delmarva Power and Light Co. | New Jersey | 1977 | 1,090 | 4 |

Table 1.4-1

WESTINGHOUSE PRESSURIZED WATER REACTOR NUCLEAR POWER PLANTS (Sheet 6 of 15)

| Plant | Owner Utility | Location | Scheduled Commercial Operation | MWe Net | Number of Loops |
|---|---|---|---|---|---|
| Sequoyah No. 1 | Tennessee Valley Authority | Tennessee | 1980 | 1,148 | 4 |
| Almaraz No. 2 | Union Electrica, S. A.; Compania Sevillana de Electricidad, S. A.; Hidroelectrica Espanola, S. A. | Spain | 1983 | 902 | 3 |
| Angra dos Reis | Furnas-Centrais Electricas, S. A. | Brazil | 1981 | 626 | 2 |
| Asco No. 1 | Fuerzas Electricas de Cataluna, S. A. (FECSA) | Spain | 1984 | 902 | 3 |
| Diablo Canyon No. 2 | Pacific Gas and Electric Co. | California | 1982 | 1,106 | 4 |
| Joseph M. Farley No. 2 | Alabama Power Company | Alabama | 1981 | 829 | 3 |
| North Anna No. 1 | Virginia Electric and Power Co. | Virginia | 1977 | 898 | 3 |
| North Anna No. 2 | Virginia Electric and Power Co. | Virginia | 1981 | 898 | 3 |

Table 1.4-1

WESTINGHOUSE PRESSURIZED WATER REACTOR NUCLEAR POWER PLANTS (Sheet 7 of 15)

| Plant | Owner Utility | Location | Scheduled Commercial Operation | MWe Net | Number of Loops |
|---|---|---|---|---|---|
| Ohi No. 1 | The Kansai Electric Power Company, Inc. | Japan | 1977 | 1,122 | 4 |
| Ohi No. 2 | The Kansai Electric Power Company, Inc. | Japan | 1979 | 1,122 | 4 |
| Ringhals No. 3 | Statens Vattenfallsverk (SSPB) | Sweden | 1981 | 900 | 3 |
| Sequoyah No. 2 | Tennessee Valley Authority | Tennessee | 1981 | 1,148 | 4 |
| Asco No. 2 | Fuerzas Electricas de Cataluna, S. A. (FECSA); Empresa Nacional Hidroelectrica del Ribagorzana, S. A. (ENHER); Fuerzas Hidroelectricas del Segre S. A.; Hidro-electrica de Cataluna, S. A. | Spain | 1985 | 902 | 3 |
| Krsko | Savske Elektrarne Ljubljana; Slovenia; Electroprivreda Zagreb, Croatia | Yugoslavia | 1981 | 615 | 2 |

Table 1.4-1

WESTINGHOUSE PRESSURIZED WATER REACTOR NUCLEAR POWER PLANTS (Sheet 8 of 15)

| Plant | Owner Utility | Location | Scheduled Commercial Operation | MWe Net | Number of Loops |
|---|---|---|---|---|---|
| Lemoniz No. 2 | Iberduero, S. A. | Spain | 1981 | 902 | 3 |
| Watts Bar No. 1 | Tennessee Valley Authority | Tennessee | 1982 | 1,177 | 4 |
| William B. McGuire No. 1 | Duke Power Company | North Carolina | 1981 | 1,180 | 4 |
| Millstone No. 3 | Northeast Nuclear Energy Co. | Connecticut | 1986 | 1,156 | 4 |
| Ringhals No. 4 | Statens Vattenfallsverk (SSPB) | Sweden | 1982 | 900 | 3 |
| Salem No. 2 | Public Service Electric and Gas Company; Philadelphia Electric Co.; Atlantic Electric Co.; Delmarva Power and Light Co. | New Jersey | 1981 | 1,115 | 4 |
| Virgil C. Summer | South Carolina Electric and Gas Company | South Carolina | 1982 | 900 | 3 |
| Watts Bar No. 2 | Tennessee Valley Authority | Tennessee | 1983 | 1,177 | 4 |

Table 1.4-1

WESTINGHOUSE PRESSURIZED WATER REACTOR NUCLEAR POWER PLANTS (Sheet 9 of 15)

| Plant | Owner Utility | Location | Scheduled Commercial Operation | MWe Net | Number of Loops |
|---|---|---|---|---|---|
| William B. McGuire No. 2 | Duke Power Company | North Carolina | 1982 | 1,180 | 4 |
| Byron No. 1 | Commonwealth Edison Co. | Illinois | 1983 | 1,120 | 4 |
| Catawba No. 1 | Duke Power Company | South Carolina | 1983 | 1,153 | 4 |
| Comanche Peak No. 1 | Texas Utilities Generating Co. | Texas | 1982 | 1,150 | 4 |
| Kori No. 2 | Korea Hydro & Nuclear Power Co. | Korea | 1983 | 605 | 2 |
| Seabrook No. 1 | Public Service Company of New Hampshire; United Illuminating Company | New Hampshire | 1983 | 1,200 | 4 |
| South Texas Project Unit No. 1 | Houston Lighting and Power Co.; Central Power and Light Co.; City Public Service of San Antonio; City of Austin, Texas | Texas | 1984 | 1,250 | 4 |
| Sayago No. 1 | Iberduero, S. A. | Spain | 1986 | 1,000 | 3 |

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Table 1.4-1

WESTINGHOUSE PRESSURIZED WATER REACTOR NUCLEAR POWER PLANTS (Sheet 10 of 15)

| Plant | Owner Utility | Location | Scheduled Commercial Operation | MWe Net | Number of Loops |
|---|---|---|---|---|---|
| Vandellos No. 2 | Electrica de Cataluna S. A.; Hidroelectrica de Cataluna S. A.; Fuerzas Electricas Del Segre | Spain | 1986 | | |
| Beaver Valley No. 2 | Duquesne Light Company; Ohio Edison Company; Pennsylvania Power Co.; Cleveland Electric Illuminating Company; Toledo Edison Company | Pennsylvania | 1987 | 852 | 3 |
| Braidwood No. 1 | Commonwealth Edison Company | Illinois | 1985 | 1,120 | 4 |
| Callaway No. 1 | SNUPPS - Union Electric Co. | Missouri | 1982 | 1.150 | 4 |
| Jamesport No. 1 | Long Island Lighting Company; New York State Electric and Gas Corp. | New York | (5) | 1,150 | 4 |
| NORCE (Aguirre) | Puerto Rico Water Resources Authority | Puerto Rico | (5) | 583 | 2 |

Table 1.4-1

WESTINGHOUSE PRESSURIZED WATER REACTOR NUCLEAR POWER PLANTS (Sheet 11 of 15)

| Plant | Owner Utility | Location | Scheduled Commercial Operation | MWe Net | Number of Loops |
|---|---|---|---|---|---|
| Seabrook No. 2 | Public Service Company New Hampshire; United Illuminating Company | New Hampshire | 1984 | 1,200 | 4 |
| Braidwood No. 2 | Commonwealth Edison Company | Illinois | 1986 | 1,120 | 4 |
| Byron No. 2 | Commonwealth Edison Company | Illinois | 1984 | 1,120 | 4 |
| Catawba No. 2 | Duke Power Company | South Carolina | 1984 | 1,153 | 4 |
| Comanche Peak No. 2 | Texas Utility Generating Co. | Texas | 1983 | 1,150 | 4 |
| South Texas Project Unit No. 2 | Houston Lighting and Power Co.; Central Power and Light Co.; City Public Service of San Antonio; City of Austin, Texas | Texas | 1984 | 1,250 | 4 |

Table 1.4-1

WESTINGHOUSE PRESSURIZED WATER REACTOR NUCLEAR POWER PLANTS (Sheet 12 of 15)

| Plant | Owner Utility | Location | Scheduled Commercial Operation | MWe Net | Number of Loops |
|---|---|---|---|---|---|
| Sterling | SNUPPS - Rochester Gas and Electric Corporation; Central Hudson Gas and Electric Corporation; Niagara Mohawk Power Corporation; Orange and Rockland Utilities, Inc. | New York | (5) | 1,150 | 4 |
| Maanshan No. 1 | Taiwan Power Company | Taiwan | 1983 | 892 | 3 |
| Wolf Creek Unit No. 1 | SNUPPS - Kansas Gas and Electric Company; Kansas City Power and Light Company | Kansas | 1983 | 1,183 | 4 |
| Alvin W. Vogtle No. 1 | Georgia Power Company; Oglethorpe Electric Membership Corp., Municipal Authority of Georgia; City of Dalton, Georgia | Georgia | 1985 | 1,113 | 4 |

Table 1.4-1

WESTINGHOUSE PRESSURIZED WATER REACTOR NUCLEAR POWER PLANTS (Sheet 13 of 15)

| Plant | Owner Utility | Location | Scheduled Commercial Operation | MWe Net | Number of Loops |
|---|---|---|---|---|---|
| Alvin W. Vogtle No. 2 | Georgia Power Company Oglethorpe Electric Membership Corp., Municipal Authority Georgia; City of Dalton, Georgia | Georgia | 1987 | 1,113 | 4 |
| Callaway No. 2 | SNUPPS - Union Electric Company | Missouri | 1988 | 1,150 | 4 |
| NEP-1 | New England Power Company | Maine | (5) | 1,150 | 4 |
| Fort Calhoun No. 2 | Omaha Public Power District; Nebraska Public Power District | Nebraska | (5) | 1,136 | 4 |
| Jamesport No. 2 | Long Island Lighting Company; New York State Electric and Gas Corp. | New York | (5) | 1,150 | 4 |
| Haven No. 1 | Wisconsin Electric Power Co.; Wisconsin Power and Light Co.; Wisconsin Public Service Corp. | Wisconsin | (5) | 900 | 3 |

Table 1.4-1

WESTINGHOUSE PRESSURIZED WATER REACTOR NUCLEAR POWER PLANTS (Sheet 14 of 15)

| Plant | Owner Utility | Location | Scheduled Commercial Operation | MWe Net | Number of Loops |
|---|---|---|---|---|---|
| Marble Hill No. 1 | Public Service Company of Indiana, Inc.; Northern Indiana Public Service Company; East Kentucky Power Corp. Inc.; Wabash Valley Power Association | Indiana | 1984 | 1,130 | 4 |
| Sears Island | Central Maine Power Company | Maine | (5) | 1,200 | 4 |
| Maanshan No. 2 | Taiwan Power Company | Taiwan | 1984 | 892 | 3 |
| Marble Hill No. 2 | Public Service Company of Indiana, Inc.; Northern Indiana Public Service Company; East Kentucky Power Assoc. | Indiana | 1985 | 1,130 | 4 |
| Shearon Harris No. 1 | Carolina Power and Light Co. | North Carolina | 1984 | 900 | 3 |
| NEP-2 | New England Power Company | Maine | (5) | 1,150 | 4 |
| Shearon Harris No. 2 | Carolina Power and Light Co. | North Carolina | 1987 | 900 | 3 |

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Table 1.4-1

WESTINGHOUSE PRESSURIZED WATER REACTOR NUCLEAR POWER PLANTS (Sheet 15 of 15)

| Plant | Owner Utility | Location | Scheduled Commercial Operation | MWe Net | Number of Loops |
|---|---|---|---|---|---|
| Shearon Harris No. 4 | Carolina Power and Light Co. | North Carolina | 1989 | 900 | 3 |
| Shearon Harris No. 3 | Carolina Power and Light Co. | North Carolina | 1991 | 900 | 3 |
| Napot Point No. 1 | National Power Corp. | Philippines | 1985 | 620 | 2 |
| Kori Unit 3 | Korea Hydro & Nuclear Power Co. | Korea | 1984 | 990 | 3 |
| Kori Unit 4 | Korea Hydro & Nuclear Power Co. | Korea | 1985 | 990 | 3 |
| Hanbit Unit 1 | Korea Hydro & Nuclear Power Co. | Korea | 1986 | 990 | 3 |
| Hanbit Unit 2 | Korea Hydro & Nuclear Power Co. | Korea | 1987 | 990 | 3 |

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Table 1.4-2

OPERATIONAL EXPERIENCE WITH UNITS DESIGNED AND CONSTRUCTED BY GEC TURBINE GENERATORS LTD

| Authority | Installation | Reactor | Turbine Generator Rating MW | Speed rpm | TSV press Mpa gauge | Service Year | Operational Hours |
|---|---|---|---|---|---|---|---|
| HEPC of Ontario | Douglas Point | PHWR | 220 | 1800 | 4.0 | 1967 | 43,000 app |
| CEGB | Winfrith Heath | SGHWR | 100 | 3000 | 6.3 | 1967 | 42,000 |
| | Dodewaard (Manufactured under license by Stork Rotterdam) | BWR(G.E.) | 54 | 3000 | 6.6 | 1968 | 62,320 |
| Atomic Energy Commission of India | Rajasthan 1 (R.A.P.F. 1) | PHWR | 220 | 3000 | 4.0 | 1972 | 15,000 |
| Swedish State Power Board | Ringhals 1 | BWR(ASEA-Atom) | 400 | 3000 | 6.7 | 1975 | 11,000 |
| Swedish State Power Board | Ringhals 2 | BWR(ASEA-Atom) | 400 | 3000 | 6.7 | 1975 | 7,000 |
| Korea Electric Co. | Ko-Ri 1 | PWR(WECo) | 600 | 1800 | 5.3 | 1977 | 35,000 app |

Table 1.4-3

UNITS GEC TURBINE GENERATORS LTD IS CURRENTLY SUPPLYING (Sheet 1 of 2)

| Authority | Installation | Reactor | Turbine Generator Rating MW | Speed rpm | TSV Press MPa Gauge | Service Year |
|---|---|---|---|---|---|---|
| Atomic Energy Commission of India | Rajasthan (RAPP 2) | PHWR | 220 | 3000 | 4.0 | 1977 |
| Atomic Energy Commission of India | Kalpakkam 1 (MAPP 1) | PHWR | 236 | 3000 | 4.0 | 1977 |
| Atomic Energy Commission of India | Kalpakkam 2 (MAPP 2) | PHWR | 236 | 3000 | 4.0 | 1979 |
| Southern California Edison Co | San Onofre 2 | PWR(Comb Eng) | 1180 | 1800 | 5.9 | 1980 |
| Detroit Edison | Enrico Fermi | BWR(G.E.) | 1280 | 1800 | 6.7 | 1980 |
| Southern California Edison Co | San Onofre 3 | PWR(Comb Eng) | 1180 | 1800 | 5.9 | 1981 |
| 554 Korea Hydro & Nuclear Power Co. | Ko-Ri 2 | PWR(WECo.) | 650 | 1800 | 6.1 | 1983 |

Table 1.4-3

UNITS GEC TURBINE GENERATORS LTD IS CURRENTLY SUPPLYING (Sheet 2 of 2)

| Authority | Installation | Reactor | Turbine Generator Rating MW | Speed rpm | TSV Press MPa Gauge | Service Year |
|---|---|---|---|---|---|---|
| Atomic Energy Commission of India | Narora 1) major portion | PHWR | 236 | 3000 | 4.0 | 1980 |
| Atomic Energy Commission of India | Narora 2) manuf under license by HEIL India | PHWR | 236 | 3000 | 4.0 | 1981 |

NOTES: From the contracts mentioned, GEC has engineered and built plants for water cooled reactors having:

1. Suitable physical and control interfaces and also appropriate control equipment required for safe and efficient operation of reactors and turbines

2. Specialized equipment for moisture removal and reheating of wet steam and for control of drain disposal, or reheat temperature, etc.

3. Specialized features for protection of the high pressure turbine horizontal joint and other sealing faces against erosion by any leaking wet stream

4. Features for protection of the turbine generator against overspeed by steam stored in the separator/reheater vessels.

5. These plants are cancelled.

Table 1.4-4

HP TURBINE COMPARATIVE DESIGN RATING PARAMETERS

| Design Rating Parameters | | KNU 5 & 6 | KORI 1 | RINGHALS 1 and 2 | SAN ONOFRE 2 and 3 |
|---|---|---|---|---|---|
| | System Frequency | 60 Hz | 60 Hz | 50 Hz | 60 Hz |
| | Rotational Speed | 1800 rpm | 1800 rpm | 3000 rpm | 1800 rpm |
| | Units | | | | |
| P - HP turbine inlet pressure | psia | 909 | 733 | 915 | 778 |
| $(\frac{Gv}{N})$ - volumetric flow parameter | cu ft/s | 840 | 621 | 540 | 1105 |
| $T_i$ - average torque developed in each HP turbine stage | lb ft | 98722 | 52767 | 61900 | 103287 |
| 1st stage mean dia<br>1st stage blade height<br>last stage blade height | in.<br>in.<br>in. | 62.94<br>4.0<br>11.0 | 66.29<br>3.13<br>7.29 | 49.04<br>2.94<br>10.38 | 64.1<br>4.8<br>12.1 |
| S - sealing duty coefficient | lb/in. | 41300 | 43600 | 52000 | 46600 |

Table 1.4-5

COMPARISON OF PRINCIPAL PHYSICAL DIMENSIONS AND WEIGHTS
OF SIMILAR LARGE CYLINDERS

HP TURBINE

| | Units | KNU 5 & 6 | KORI 1 | PETERHEAD IP | SAN ONOFRE 2 AND 3 |
|---|---|---|---|---|---|
| Total weight of turbine cylinder | lb | 502,000 | 504,000 | 340,480 | 504,000 |
| Rotor weight | lb | 128,000 | 128,000 | 71,680 | 128,000 |
| Rotor bearing span | in. | 259 | 259 | 175 | 259 |
| Rotor maximum diameter | in. | 58.0 | 59.0 | 57.5 | 58.0 |
| Max. rotor bearing diameter | in. | 30.0 | 30.0 | 20.0 | 30.0 |
| Max. casing diameter | in. | 138.0 | 140.0 | 130.0 | 140.0 |

Table 1.4-6

COMPARISON OF PRINCIPAL BLADING PARAMETERS AND OF BLADE AND ROTOR STRESSING OF SIMILAR LARGE CYLINDERS

| | Units | KNU 5 & 6 | KORI 1 | PETERHEAD IP | SAN ONOFRE 2 AND 3 |
|---|---|---|---|---|---|
| First stage blade height | in. | 4.0 | 3.13 | 2.70 | 4.83 |
| Last stage blade height | in. | 11.0 | 7.29 | 8.07 | 12.10 |
| Centrifugal rotor stress first stage | $lb/in.^2$ | 7,700 | 7,652 | 13,440 | 7,750 |
| Centrifugal rotor stress last stage | $lb/in.^2$ | 9,000 | 8,920 | 19,264 | 9,200 |
| Centrifugal blade stress first stage | $lb/in.^2$ | 3,584 | 2,464 | 6,496 | 3,584 |
| Centrifugal blade stress last stage | $lb/in.^2$ | 9,184 | 4,480 | 12,096 | 6,496 |
| Operating temperature first stage | $^oC(^oF)$ | 278(533) | 264(508) | 538(1,000) | 271(520) |
| Operating temperature last stage | $^oC(^oF)$ | 193(380) | 199(390) | 260(500) | 202(395) |

Table 1.4-7

LP TURBINE COMPARATIVE DESIGN RATING PARAMETERS (Sheet 1 of 2)

| Design Rating Parameters | Units | KNU 5 & 6 | KORI 1 | RINGHALS[a] 1 and 2 | SAN ONOFRE 2 and 3 |
|---|---|---|---|---|---|
| G - last stage steam flow | lbs/hr | 2,281,532 | 2,032,224 | 892,152 | 2,494,237 |
| Ae - last stage annulus area (2 flows) | $ft^2$ | 265 | 265 | 155.4 | 265 |
| $\left(\frac{G}{Ae}\right)$ - exhaust blade loading | lbs/hr $ft^2$ | 8,532 | 7,669 | 5,741 | 9,412 |
| x - wetness at inlet to last stage | % | 8.1 | 7.5 | 7.0 | 8.1 |
| p - pressure before last stage (tip) | psia | 2.1 | 1.8 | 1.53 | 2.4 |
| c - velocity at discharge from last stationary blade | ft/s | 900 | 900 | | 900 |

Table 1.4-7

LP TURBINE COMPARATIVE DESIGN RATING PARAMETERS (Sheet 2 of 2)

| Design Rating Parameters | Units | KNU 5 & 6 | KORI 1 | RINGHALS[a] 1 and 2 | SAN ONOFRE 2 and 3 |
|---|---|---|---|---|---|
| $G \dfrac{x}{p^2 \cdot c\,\beta}$ relative erosion criterion | Relative to Ko-Ri 1 | 0.89 | 1.00 | | 0.75 |

a. Ringhals has a 36" last-stage blade running at 3000 rpm and is therefore not similar to the other installations. It is, however, constructed in accordance with the same basic design concepts and operates satisfactorily at substantially higher tip speeds and stresses.

Table 1.4-8

COMPARISON OF PRINCIPAL PHYSICAL DIMENSIONS AND WEIGHTS

LP TURBINES

| | Units | KNU 5 & 6 | KORI 1 | RINGHALS 1 AND 2 | SAN ONOFRE 2 AND 3 |
|---|---|---|---|---|---|
| Rotor weight | lb. | 369,600 | 369,600 | 107,520 | 369,600 |
| Overall rotor length | in. | 419.5 | 419.5 | 299.5 | 419.5 |
| Rotor maximum diameter | in. | 90 | 90 | 64 | 90 |
| Max bearing diameter | in. | 30 | 30 | 19 | 30 |
| Max casing diameter | in. | 220 | 220 | 146 | 220 |

Table 1.4-9

COMPARISON OF PRINCIPAL BLADING PARAMETERS

AND OF BLADE AND ROTOR STRESSING

| | Units | KNU 5 & 6 | KORI 1 | RINGHALS 1 AND 2 | SAN ONOFRE 2 AND 3 |
|---|---|---|---|---|---|
| First stage blade height | in. | 3.0 | 2.5 | 1.7 | 3.0 |
| Last stage blade height | in. | 45.0 | 45.0 | 36.0 | 45.0 |
| Centrifugal rotor stress | | | | | |
| first stage | $lb/in.^2$ | 43,000 | 42,744 | 36,064 | 43,000 |
| last stage | $lb/in.^2$ | 49,428 | 49,428 | 39,200 | 49,428 |
| Centrifugal blade stress | | | | | |
| first stage | $lb/in.^2$ | 4,256 | 3,808 | 5,376 | 4,480 |
| last stage | $lb/in.^2$ | 41,440 | 41,440 | 67,872 | 41,440 |
| Operating temperature | | | | | |
| first stage | $^{o}C(^{o}F)$ | 267(512) | 251(484) | 266(510) | 256(493) |
| last stage | $^{o}C(^{o}F)$ | vacuum | vacuum | vacuum | vacuum |
| Wetness before last stage | % | 8.1 | 7.5 | 7.0 | 8.1 |
| Wetness after last stage | % | 11.8 | 11.4 | 10.8 | 11.8 |

Table 1.4-10

SEPARATOR/REHEATER COMPARATIVE DESIGN RATING PARAMETERS

| Design Rating Parameter | Units | KNU 5 and 6 | SAN ONOFRE 2 and 3 |
|---|---|---|---|
| G - mass flow of main cycle steam (from HP turbine exhaust) | lbs/hr | 10,293,177 | 10,937,329 |
| x - inlet wetness of steam | % | 12.8 | 11.8 |
| p - pressure of main steam | psia | 195 | 173 |
| T - temperature to which steam is reheated | $^{o}C$ ($^{o}F$) | 267 (512.6) | 257 (495) |
| $\Delta T$- (live steam temperature - T) | $^{o}C$ ($^{o}F$) | 14 (25) | 14 (25) |
| A - total reheater heat exchange surface | $ft^2$ | 260,000 | 468,000 |

Table 1.4-11

SEPARATOR/REHEATER COMPARATIVE DESIGN PARAMETERS

| | KNU 5 & 6 | SAN ONOFRE | RINGHALS |
|---|---|---|---|
| No. Vessels | 2 | 2 | 2 |
| Overall length, m | 26.2 (85.96') | 37.2 | 12.8 |
| Shell diameter, mm | 4115 (13.5') | 4115 | 3658 |
| Shell thickness, mm | 38 (1.5") | 38 | 32 |
| Total tube length, m | | | |
| Live steam | 41,400 (135.833') | 51,646 | 25,346 |
| Bled steam | 24,330 (79,827') | 67,073 | - |
| Tube surface $m^2/m$ (1.20 $ft^2/ft$) | 0.366 | 0.366 | 0.247 |
| Tube material | M.S. | M.S. | S.S. |
| Tubeplate material | M.S. | M.S. | M.S. (S.S. clad) |
| Design pressure, kPa | 1,450 (210.3 psi) | 1,379 | 1,379 |
| Design temperature, $^oC$ | 205 (401F) | 198 | 198 |
| Design code | BS1500 | ASME VIII Div. 1 | Swedish Stds and ASME VIII |

Table 1.4-12

PRINCIPAL GENERATOR DESIGN PARAMETERS

| Parameter | units | KRN 1 | KRN 3 & 4 | ENRICO FERMI | |
|---|---|---|---|---|---|
| Output | MW/MVA | 595/700 | 1,100/1,222 | 1,215/1,350 | |
| Voltage | kV | 22 | 22 | 22 | |
| Frequency | Hz | 60 | 60 | 60 | |
| Speed | rpm | 1,800 | 1,800 | 1,800 | |
| Rotor body diameter | mm | 1,651 | 1,708.2 | 1,651 | 433 |
| Rotor body length | mm | 5,182 | 7,500 | 8,407 | |
| Air gap length | mm | 83.8 | 120.9 | 71.1 | |
| No. of stator slots | | 66 | 72 | 72 | 390 |
| Current/ conductor | A | 9,185 | 10,692 | 8,857 | |
| Stator current density | $A/mm^2$ | LNB 9.48<br>LFB 8.01 | LNB 8.70<br>LFB 7.10 | LNB 11.14<br>LFB 9.56 | |
| Rotor turns/ pole | | 43 | 46 | 43 | |
| Rotor CMR current | A | 4,715 | 5,436 | 5,095 | |
| Rotor current Density | $A/mm^2$ | 6.85 | 7.81 | 7.40 | |
| stator slot width | mm | 38.4 | 43.4 | 33.1 | |
| Bar bumping force | kg/mm | 1.13 | 1.3 | 1.21 | |
| Peak airgap flux density | Tesla | 1.096 | 0.92 | 1.12 | |
| Electric loading | ac/m. | 212,760 | 251,322 | 226,564 | |

# 1.5

## 1.5 REQUIREMENTS FOR FURTHER TECHNICAL INFORMATION

This design is based upon proven concepts which have been developed and successfully applied to the design of pressurized water reactor systems. There are currently no areas of research and development which are required for operation of this plant.

Reference 1 presents descriptions of those safety-related Research and Development Programs which have been carried out for, by, or in conjunction with Westinghouse Nuclear Energy Systems, and which are applicable to Westinghouse pressurized water reactors.

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1.6

## 1.6 MATERIAL INCORPORATED BY REFERENCE

### 1.6.1 WESTINGHOUSE TOPICAL REPORTS

This section lists those Westinghouse topical reports, referenced throughout this FSAR, which provide NRC-required information additional to that provided in Table 1.6-1. These reports have been filed separately with the NRC in support of similar applications.

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### 1.6.2 BECHTEL TOPICAL REPORTS

Bechtel Topical Reports incorporated by references are listed in Table 1.6-1.

Table 1.6-1

BECHTEL TOPICAL REPORTS (Sheet 1 of 2)

| Bechtel Topical Report No. | Title | Revision Number | Section Reference | USNRC Approval Status |
|---|---|---|---|---|
| BC-TOP-1 | Containment building liner plate design | 1 | 3.8.1 | Approved with conditions 2-7-74 |
| BC-TOP-3-A | Tornado and extreme wind design criteria for nuclear power plants | 3 | 3.3<br>3.5<br>3.8.1 | Approved 10-4-74 |
| BC-TOP-4-A | Seismic analyses of structures and equipment for nuclear power plants | 3 | 3.7<br>3.8.1 | Approved 10-31-74 |
| BC-TOP-5-A | Prestressed concrete nuclear reactor containment structures | 3 | 3.8.1<br>3A/1.18<br>6.2.1 | Approved 3-28-75 |
| BC-TOP-7 | Full-scale buttress test for prestressed nuclear containment structures | 0 | 3.8.1<br>3A/1.18 | Approved 8-24-73 |
| BC-TOP-8 | Tendon end anchor reinforcement test | 0 | 3.8.1<br>3A/1.18 | Approved 8-24-73 |
| BC-TOP-9-A | Design of structures for missile impact | 2 | 3.5<br>3.6.5<br>3.8.1 | Approved 11-25-74 |

Table 1.6-1

BECHTEL TOPICAL REPORTS (Sheet 2 of 2)

| Bechtel Topical Report No. | Title | Revision Number | Section Reference | USNRC Approval Status |
|---|---|---|---|---|
| BN-TOP-1 | Testing criteria for integrated leak rate testing of primary containment structures for nuclear power plants | 1 | 3.8.1<br>6.2.1 | Approved<br>2-1-73 |
| BN-TOP-2 | Design for pipe break effects | 2 | 3.6<br>3.8.1<br>3A/1.46 | Approved<br>6-17-74 |
| BN-TOP-3 | Performance and sizing of dry pressure containments | 3 | 3.8.1<br>6.2.1 | Submitted<br>8-6-75 |
| BN-TOP-4 | Subcompartment pressure and temperature transient analysis | 1 | 3.6 | Approved<br>2-23-79 |
| BP-TOP-1 | Seismic analysis of piping systems | 3 | 3.7<br>3.9 | Approved<br>9-29-76 |
| BQ-TOP-1 | Bechtel quality assurance program for nuclear power plants | 2A | Refer to a Separately Published QA Manual | Approved<br>July 1977 |

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## 1.7 DRAWINGS AND OTHER DETAILED INFORMATION

The figures included in each chapter of FSAR are reference only and the drawings listed in Table 1.7-1 and 1.7-2 are only controlled drawings.
The drawing list contains FSAR figure & section number, drawing type, controlled drawing number, title, revision number and revision date. 539
However, if there is no applicable controlled drawing, the figure included in FSAR shall be deemed controlled drawing.

### 1.7.1 ELECTRICAL, INSTRUMENTATION AND CONTROL DRAWINGS

Table 1.7-1 contains a list of electrical, instrumentation controll drawings and the corresponding FSAR figure number if applicable. These drawings are considered necessary to evaluate the safety-reated features of KRN 3&4. 539

### 1.7.2 PIPING AND INSTRUMENTATION DIAGRAMS

Table 1.7-2 contains a list of piping and instrumentation diagram and the corresponding FSAR figure number if applicable.
The p&id legend, figures 1.1-1 and 1.1-2, provides an explanation of symbols and characters used in these FSAR figures.

### 1.7.3 OTHER DETAILED INFORMATION

Information will be provided as requested by ROK-NRB. 539

## 1.8 Improvement Action Items For Post Fukushima Daiichi Accident 475

Improvement action items issued in the domestic NPP safety review report performed as a part of countermeasures post Fukushima Daiichi accident are described in Appendix 1A.