Amendment 349
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YGN 3&4 FSAR

CHAPTER 1 - INTRODUCTION AND GENERAL PLANT DESCRIPTION

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CHAPTER 1 INTRODUCTION AND GENERAL PLANT DESCRIPTION

1.1 INTRODUCTION

This Final Safety Analysis Report is submitted to Nuclear Safety And Security Commission (NSSC) in support of an application for an operating license. The 582
application is made by the Korea Electric Power Corporation for HANBIT Nuclear Power Plant[1], Units 3 & 4 (HANBIT 3&4)[2], two identical units, to 658
be located at [redacted]

1.1.1 Yonggwang Nuclear Power Plant, Units 3 & 4

The plant site is located on the southwestern coast of the Korean Peninsula approximately [redacted]

Kwangju City is the nearest population center of 25,000 or more. It has a population of 1,144,695. Other population centers of 25,000 or more within 50 miles (80 km) from the site are Kunsan, approximately 40 miles (64 km) north-northeast of the site, with a population of 218,216; Chonju, approximately 46 miles (73 km) northeast of the site, with a population of 517,104; and Mokpo, approximately 40 miles (64 km) south of the site, with a population of 253,423.

The two-loop pressurized water reactor nuclear steam supply systems are provided by Korea Heavy Industries & Construction Co., Ltd. (KHIC), Korea Atomic Energy Research Institute (KAERI), and ABB-Combustion Engineering, Inc. (ABB-CE). The nuclear steam supply systems each is housed in a prestressed concrete, steel-lined containment designed by Korea Power Engineering Company, Inc. (KOPEC) and Sargent & Lundy Engineers (S&L).

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1) On May 7, 2013, we changed the name of the Site in accordance with Organization Regulation NO.647 of KHNP. Therefore, on the following FSAR, the 'Yonggwang Nuclear Power Plant' is regarded as 'HANBIT Nuclear Power Plant'. 658
2) And 'YGN' is regarded as 'HANBIT'.

The reactor power levels and corresponding net electrical power outputs are listed in Table 1.1-1.

The major construction milestones are listed in Table 1.1-2.

The nuclear steam supply system (NSSS) is designed for an NSSS output of 2825 MWt, which is defined as the rated power in the application.

The turbine-generator is rated for operation at the NSSS output of 2825 MWt with a corresponding electrical output of 1092 MWe. This corresponds to a guaranteed generator output for each unit of 1049 MWe. The turbine-generator is supplied by KHIC and General Electric Company.

1.1.2 The Final Safety Analysis Report

1.1.2.1 Organization and Format

This FSAR for YGN 3&4 follows Regulatory Guide 1.70, Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants, Revision 3, prepared by the Regulatory Staff of the U. S. Nuclear Regulatory Commission (USNRC), issued in November 1978. Chapter 16, Technical Specifications, can be replaced with Improved Technical Specification subject to MOST approval. 1 Chapter 18, Human Factors Engineering, is also included, although it is not required by Regulatory Guide 1.70, Rev.3.

The FSAR is paginated to provide flexibility when incorporating changes to text and figures. All text pages are numbered by section except Chapter 15; e.g., 1.1-1 is the first page of Section 1.1. Tables and figures are 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. Due to the manner in which material is organized in Chapter 15, the pages, tables, and figures are numbered by subsection. In some cases, appendices are included at the end of a chapter of the FSAR to provide supplemental information.

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.

Many parameters in the FSAR text are given in English units (ft-lb); however, metric units are also provided where appropriate for the benefit of those who may be more familiar with the metric system. The metric parameters given are often approximate (i.e., rounded-off) and should be used for information only.

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

1.1.2.2 Drawings

Drawings are included throughout this FSAR with the appropriate system descriptions. Figures listed in Table 1.7-1,1.7-2 and 1.7-3 are typical and the only controlled drawings are actually used to operate the plant. Symbols and abbreviations used on flow diagrams and piping and instrumentation diagrams (P&IDs) are shown in Figure 1.1-1. 499

1.1.2.3 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
b. ITS Chapter 1 Table 1.1-1 is 운영기술지침서 제1편 표 1.1-1

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

a. ITS Base B 3.4 is 운영기술지침서 기술배경서 B 3.4

TABLE 1.1-1

POWER LEVELS

Type of Power	Approximate Rated and Design Power Level
Core thermal power level (MWt)	2815
Reactor coolant pump work (MWt)	10
Gross electrical power output at generator terminals at valves wide open (MWe)	1092
Main turbine-generator output, guaranteed (MWe)	1049

TABLE 1.1-2

MAJOR CONSTRUCTION MILESTONES

Milestone	Date Unit 3	Date Unit 4
Authorization to proceed	Apr. 30, 87	Apr. 30, 87
Submit PSAR	Mar. 31, 88	Mar. 31, 88
Construction permit	Dec. 21, 89	Dec. 21, 89
First concrete	Dec. 23, 89	May 26, 90
Submit FSAR	Dec. 31, 92	Dec. 31, 92
Cold hydro test	Dec. 01, 93	Dec. 01, 94
Hot functional test	Apr. 01, 94	Apr. 01, 95
Fuel load	Aug. 01, 94	Aug. 01, 95
Commercial operation	Mar. 31, 95	Mar. 31, 96

KOREA ELECTRIC POWER CORPORATION
YONGGWANG 3 & 4
FSAR

P & I DIAGRAM
SYMBOLS AND LEGEND
(Sheet 1 of 5)

Figure 1.1-1

KOREA ELECTRIC POWER CORPORATION
YONGGWANG 3 & 4
FSAR

P & I DIAGRAM
SYMBOLS AND LEGEND
(Sheet 2 of 5)

Figure 1.1-1

KOREA ELECTRIC POWER CORPORATION
YONGGWANG 3 & 4
FSAR

P & I DIAGRAM
SYMBOLS AND LEGEND
(Sheet 3 of 5)

Figure 1.1-1

KOREA ELECTRIC POWER CORPORATION
YONGGWANG 3 & 4
FSAR

P & I DIAGRAM
SYMBOLS AND LEGEND
(Sheet 4 of 5)

Figure 1.1-1

KOREA ELECTRIC POWER CORPORATION
YONGGWANG 3 & 4
FSAR

P & I DIAGRAM
SYMBOLS AND LEGEND
(Sheet 5 of 5)

Figure 1.1-1

1.2 GENERAL PLANT DESCRIPTION

1.2.1 Site Description

1.2.1.1 Site Location

The HANBIT site[1] is on the southwestern coast of the Republic of Korea near 658
the town of [redacted]

The nearest airport to the site is the Kwangju Airport, at Kwangju, 60 km southeast of the site.

The sea area adjacent to the Yonggwang site is an open ocean. The site has existing docking facilities that can accommodate barges to 3000-ton capacity. It is anticipated that major equipment has been delivered by barge.

The Yonggwang site on Regional Route 842, which is approximately 9 km west of National Route 22, is the only national road passing through Yonggwang-up. Route 842 provides the only land access to the Yonggwang site.

The Honam railway line, Seoul to Mokpo via Kwangju, passes Songjong-ri Railway Station, about 55 km southeast, allowing freight service for the area. At its closest approach, this line passes about 55 km southeast of the Yonggwang site.

The site is located on the coast of Yellow Sea. The Yellow Sea has a strong semidiurnal tidal current. Its major direction is northeast during flood tide and southwest during ebb tide.

There is no history of damaging tsunamis on the west coast of Korea. The maximum calculated tsunami wave height is negligible.

1) On May 7, 2013, we changed the name of the Site in accordance with Organization Regulation NO.647 of KHNP. Therefore, on the following FSAR, the 'Yonggwang Site' is regarded as 'HANBIT Site'. 658

1.2.1.2 Geology

The Yonggwang site is located within the Hanland physiographic division of Korea. Hanland is further subdivided into three regions from north to south: Gyeonggi, Ogcheon Taebaeksan, and Gyeongsang. The Yonggwang site lies within the Ogcheon Taebaeksan subdivision and the physiography is developed upon Precambrian metasediments and granitic rocks, which were intruded during the Daebo orogeny.

Relief in the Ogcheon Taebaeksan region increases westward as the region changes from submergence to emergence. As a result of this regional trend, the Yonggwang site area, which is located on the submergent coast, consists of mountain masses dispersed within tidal lowlands. In the sense of a general overview, the central portion of the proposed site area consists of a southern mountainous area separated by an estuary from a central mountainous area, which, in turn, is separated by a second estuary from a northern area of low hills and tidal flats. The western section of the site area is composed of low hills and tidal lowland and, westward, there is a series of small, near-shore islands in the Yellow Sea.

No notable folds or major faults have been observed within the site's 8-km radius. The site area consists of two basic rock types: Cretaceous volcanics and Precambrian metamorphics and intrusives.

Geologic conditions underlying the site are known from the borings conducted at the site and the examination of the outcrops along the seacoast in the immediate vicinity of the site. The site is underlain by volcanic rocks of intermediate to acidic composition. These rocks are covered by 0 to 5 meters of soil developed in place on the underlying volcanics.

No evidence of heave suggesting unrelieved residual stresses in the bedrock has been observed, and there is no rock or soil that might be unstable because of mineralogy, physical, or chemical properties. Also, there is no withdrawal or addition of subsurface fluids or mineral extraction at the site. Refer to Section 2.5 for further information.

1.2.1.3 Meteorology

The site climatology can be described as continental in nature during the winter seasons. The strong summer weather systems are influenced primarily by tropical air from the south-southwest, while in the winter season, weather is influenced by polar continental air masses.

Snow is common during winter, with an estimated annual accumulation of 50 cm/yr and a maximum of 60 cm/yr. Precipitation during the winter season occurs principally as snow on the southwest coast.

Heavy precipitation occurs during the summer season and is most often associated with the passage of typhoons and tropical storms that occur predominately in the months of May through September.

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

The average annual rainfall is 1320 mm with 40% to 50% of the rainfall occurring during the months of June through August. The maximum hourly rainfall is 73 mm.

The annual mean temperature at Yonggwang is 13.1°C. The extreme maximum and minimum temperatures are 38°C and -19°C.

The annual mean humidity is 72.7% at Yonggwang. The high pressure that develops off the Pacific Ocean during the summer results in a moist southerly flow of air from the South Sea to the Yonggwang region. During the winter, a semipermanent high-pressure cell develops over the central region of Mongolia, resulting in a prevailing northwesterly flow of air into the Yonggwang region.

The mean annual wind speed for the Kwangju area is 1.9 m/sec. The mean annual wind speed for Chonju is 1.5 m/sec. The highest wind speed recorded at Kwangju during the 46-year period of 1938-1984 inclusive, was 32.0 m/sec from the east-southeast.

Chonju, with 40 years of records, has reported wind speeds up to 26.3 m/sec; the maximum wind speed at Chonju was recorded on August 3, 1945.

Refer to Section 2.3 for further information.

1.2.1.4 Hydrology

The Yonggwang site is surrounded by sea on two sides. The eastern sections are bordered by land.

There is no stream or river that could cause flooding of the site.

High and low seawater levels are given in Subsection 2.4.1.

1.2.1.5 Population Distribution

Population distribution is discussed in detail in Subsection 2.1.3.

1.2.2 Facility Arrangement Summary

Each power block is an identical slide-along arrangement, with the turbine building in a peninsular arrangement with respect to the containment. Each unit consists of a containment building, a primary and secondary auxiliary buildings, an access control building, a fuel building, a turbine building, a component cooling water heat exchanger building, an essential service water intake structure, and a common radwaste building.

The units are arranged to avoid interfering with the operation of YGN 1&2 or with the area reserved for two future units. The arrangement also utilizes the existing plan for the location of the intake and discharge structures. The arrangement also provides full control over personnel access through a single control point during normal operation.

See Figures 1.2-1 through 1.2-55 for the detailed arrangement of the buildings and equipment.

1.2.3 Nuclear Steam Supply System (NSSS) Summary

The NSSS generates 2825 MWt, producing saturated main steam. The NSSS contains two independent primary coolant loops, each of which has two reactor coolant pumps, a steam generator, a 42-inch (106.7-cm) ID outlet (hot) pipe and two 30-inch (76.2-cm) ID inlet (cold) pipes. An electrically heated pressurizer is connected to one of the loops, and safety injection lines are connected to each of the four cold legs and the two hot legs. Pressurized water circulates by means of electric motor-driven, single-staged, centrifugal reactor coolant pumps downward between the reactor vessel shell and the core support barrel, upward through the reactor core, through the tube side of the vertical U-tube steam generators with an integral economizer, and back to the reactor coolant pumps. The saturated steam produced in the steam generators is passed to the turbine.

1.2.3.1 Reactor Core

The reactor core is fueled with uranium dioxide pellets enclosed in zircaloy tubes with welded end caps. The tubes are fabricated into assemblies in which end fittings limit axial motion and grids limit lateral motion of the tubes. The control element assemblies (CEAs) consist of NiCrFe alloy-clad boron carbide absorber rods, which are guided by tubes located within the fuel assembly. The core consists of 177 fuel assemblies that will be initially loaded with four different U-235 enrichments. The NSSS full-thermal output is 2825 MWt with a core thermal output of 2815 MWt.

Design criteria are established to ensure the following:

a. The minimum departure from nucleate boiling ratio during normal operation and anticipated operational occurrences provides at least a 95% probability with 95% confidence that departure from nucleate boiling is not less than 1.21. 446

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. The fuel rod clad is designed to maintain cladding integrity throughout fuel life.

d. The Reactor system is designed so that any xenon transients will be adequately damped.

e. The reactor coolant system is designed and constructed to maintain its integrity throughout the expected plant life.

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, or impair operation of the engineered safety features.

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 changes in any operating variable.

The reactor core is further discussed in Chapter 4.

1.2.3.2 Reactor Pressure Vessel Internals

The internal structures are composed of the core support barrel, the lower support structure and ICI nozzle assembly, the core shroud, and the upper guide structure assembly. The core support barrel is a right circular cylinder supported by a ring flange from a ledge on the reactor vessel. It carries the entire weight of the core. Snubbers are provided at the lower end of the core support barrel to restrict radial and torsional movement. The lower support structure transmits the weight of the core to the core support barrel by means of a beam structure. The core shroud surrounds the core and minimizes the amount of bypass flow. The upper guide structure provides a flow shroud for the CEAs, and limits upward motion of the fuel assemblies.

The principal design bases for the reactor internals are to provide the vertical supports and horizontal restraints during all normal operating, upset, emergency, and faulted conditions.

The core is supported and restrained during normal operation and postulated accidents to ensure that coolant can be supplied to the coolant channels for heat removal.

Reactor internals are further discussed in Sections 3.9 and 4.5

1.2.3.3 Reactor Coolant System

The reactor coolant system (RCS) is arranged as two closed loops connected in parallel to the reactor vessel. Each loop consists of one 42-inch (106.7-cm) ID outlet (hot) pipe, one steam generator, two 30-inch (76.2-cm) ID inlet (cold) pipes, and two pumps. An electrically heated pressurizer is connected to one of the loops, and safety injection lines are connected to each of the four cold legs and two hot legs.

The RCS operates at a nominal pressure of 2250 psia (158.2 kg/cm^2 A). The reactor coolant enters near the top of the reactor vessel, then flows downward between the reactor vessel shell and the core barrel, up through the core, leaves the reactor vessel, and flows through the tube side of the two vertical U-tube steam generators with an integral economizer where heat is transferred to the main steam system. Reactor coolant pumps return the reactor coolant to the reactor vessel.

Two steam generators, using heat generated by the reactor core and carried by the primary coolant to each steam generator, produce steam for driving the plant turbine-generator. Each steam generator is a vertical U-tube heat exchanger with an integral economizer that operates with the reactor coolant on the tube side and secondary coolant on the shell side.

Each unit is designed to transfer heat from the reactor coolant system to the main steam system to produce saturated steam when provided with the proper feedwater supply. Moisture separators and steam dryers on the shell side of the steam generator limit the moisture content of the steam during normal operation at full power.

Hot reactor coolant from the reactor vessel enters the steam generator through the inlet nozzle in the primary head. It then flows through the U-tubes, where it applies heat to the secondary coolant, to the outlet side of the primary head where the flow splits and leaves through the two outlet nozzles. A vertical divider plate separates the inlet and outlet plenums of the primary head. An integral economizer is employed on the cold leg of the U-tube steam generator to enhance the generator thermal effectiveness. With fixed reactor coolant conditions, the use of an economizer enables the steam generator to operate at a higher steam pressure without an increase in heating surface.

The steam generator with an integral economizer is in most respects similar to earlier U-tube recirculating steam generators. The basic difference is that instead of introducing feedwater only through a sparger ring to mix with the recirculating water flow in the downcomer channel, feedwater is also introduced into a separate, but integral section of the steam generator. A semi-cylindrical section of the tube bundle, at the cold leg or exit end of the U-tubes, is separated from the remainder of the tube bundle by vertical divider plates. Feedwater is introduced directly into this section and preheated before discharge into the evaporator section. Feedwater flow enters the economizer through two nozzles into the distribution box. Discharge ports in the distribution box are sized and spaced to provide a uniform rate of discharge over the full half circumference of the economizer. The flow after leaving the distribution box passes radially across the tube sheet. A flow baffle acts as the upper boundary of this radial pass. This baffle is sized to evenly distribute flow through the axial region of the economizer. Flow passes upward through the annuli formed by the tubes and baffle plate into the axial flow region. This region is basically a counterflow heat exchanger, with feedwater directed upward outside the tubes and primary flow directed downward inside the tubes. Feedwater then exits the economizer slightly subcooled and enters the boiling region of the steam generator.

The steam-water mixture leaving the vertical U-tube heat transfer surface enters the separators, which impart a centrifugal motion to the mixture and separate the water particles from the steam. The water exits from the perforated separator housings and recirculates through the downcomer channel to repeat the cycle. Final drying of the steam is accomplished by passage of the steam through dryers.

An integral flow restrictor is installed in each steam-generator steam outlet nozzle. The main steam line flow restrictors are described in Subsection 5.4.4.

The reactor coolant is circulated by four electric motor-driven single-stage centrifugal pumps. The pump shafts are sealed by mechanical seals. The seal performance is monitored by pressure and temperature sensing devices in the seal system.

The RCS is further discussed in Chapter 5.

1.2.4 Principal Design Criteria

1.2.4.1 Licensing Design Basis

The plant is designed, fabricated, constructed, and operated so that any release of radioactive materials to the environment is not greater than the guideline values given in applicable NSSC and USNRC regulations. If the requirements of NSSC and USNRC regulations are in conflict, the NSSC regulations shall govern. 582

The proposed design conforms in all cases to the intent of the U.S. regulations given in 10 CFR 50, "Licensing of Production and Utilization Facilities," Appendix A, General Design Criteria for Nuclear Power Plants. Specific compliance is discussed in Section 3.1.

Principal design criteria that are incorporated in the design of the plant include, but are not limited to the following:

a. The plant is designed to maintain safe shutdown (cold shutdown) conditions in the event of a postulated design-basis accident (DBA) regardless of the worst site conditions, which are discussed in Chapter 2.

b. Any release of radioactive materials to the environment does not exceed the limitations specified by NSSC and USNRC regulations. 582

c. The plant is designed, fabricated, constructed, and operated in accordance with the applicable code of federal regulations, regulatory guides, and codes and standards of the U.S.A. and those of Korea in effect as of December 31, 1985. The applicable codes and standards include but are not limited to American Society of Mechanical Engineers (ASME), American Nuclear Society (ANS), American Society of Testing and Materials (ASTM), American National Standards Institute (ANSI), Institute of Electrical and Electronics Engineers (IEEE), and American Welding Society (AWS). The applications of these codes and standards are described in the sections where the topics related to the codes and standards are discussed.

d. To achieve the national goal of localization, Korean regulations, codes and standards, and industrial practices are taken into account to the maximum extent practical unless it adversely affects plant safety or reliability.

e. Components and structures are designed with appropriate safety margins so that a hazardous release of radioactive materials does not occur.

1.2.4.2 Severe Accident Considerations

The Individual Plant Examination (IPE) has been applied for consideration of severe accident related issues at the initiative of the utility in order to improve the plant safety.

1.2.5 ESF and Emergency Systems Summary Description

Engineered safety features (ESF) function in the highly unlikely event of an accidental release of radioactive fission products from the reactor coolant system, particularly as the result of loss-of-coolant accidents (LOCA). These safeguards function to localize, control, mitigate, or terminate such accidents to hold exposure levels below 10 CFR 100.

1.2.5.1 Containment Systems

1.2.5.1.1 Containment Spray System

The containment spray system provides a chemical additive to remove fission products, primarily elemental iodine, from the containment atmosphere for the purpose of minimizing the offsite radiological consequences following the design-basis accident. At the same time, the spray water serves to reduce containment temperature and pressure during both the injection and recirculation phases.

Two 100 % capacity spray subsystems are provided, each consisting of a pump, spray header, and spray additive components. The containment spray pumps take suction initially from a refueling water tank. The spray additive is provided from the spray additive tanks.

The containment spray system is further discussed in Subsections 6.2.2 and 6.5.2.

1.2.5.1.2 Containment Building

The containment building is a prestressed, reinforced concrete structure in the shape of a cylinder with a hemispherical dome 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. 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 % of the total containment free volume in 24 hours, even in the unlikely event of a LOCA. The internal structures and compartment arrangement provide equipment missile protection and biological shielding for maintenance personnel.

The containment building is designed for all credible loading combinations, including construction loads, test loads, normal loads, abnormal loads during a LOCA, and loads due to adverse environmental conditions as follows:

a. Dead load

b. Live load

c. Wind load

d. Loads caused by internal pressure and temperature transients due to accident conditions

e. Eexternal pressure loads

f. Thermal loads due to operating and ambient temperature

g. Post-tensioning forces

h. Liner plate expansion loads and effects

i. Polar crane support reactions

j. Missile impact loads

k. Static and dynamic earth presures

l. Hydro pressure loads

The containment building design pressure is 54 psig and 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. 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 design is discussed further in Sections 3.8 and 6.2.

1.2.5.1.3 Reactor Containment Fan Cooler System

The reactor containment fan cooler (RCFC) system is provided to cool the atmosphere during normal operation and to reduce the pressure and temperature in the containment, in conjunction with the containment spray system, following a LOCA or secondary line break. Four 50% containment fan coolers (RCFC) are provided (100% per each division).

The RCFC system is further discussed in Section 6.2.

1.2.5.2 Safety Injection System

In the highly unlikely event of a loss-of-coolant accident, the safety injection system (SIS), including high-pressure and low-pressure safety injection pumps and safety injection tanks, injects borated water into the reactor coolant system. This provides cooling to limit core damage and fission product release and ensures an adequate shutdown margin. The SIS also provides continuous long-term, postaccident cooling of the core by recirculation of borated water from the containment recirculation sump.

The SIS is discussed further in Section 6.3.

1.2.5.3 Auxiliary Power System

Two emergency diesel generators are provided for each unit and are available as onsite Class 1E power sources (in the event of a loss-of-offsite power) for operating safety-related electrical loads and selected non-safety-related loads. Each emergency diesel generator and its associated auxiliary equipment necessary for operation is physically separated and electrically isolated from the other division. Each emergency diesel generator is capable of supplying the required electrical loads for a simultaneous LOCA and loss-of-offsite power.

One diesel generator is provided for both units as an alternate AC (AAC) source to cope with station blackout (SBO). This diesel generator is capable of supplying one division of the shutdown loads of one unit.

Refer to Chapter 8 for further information on the electrical system design.

1.2.5.4 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. The fire and smoke monitoring, detection, and alarm system includes a supervisory circuit that 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.

The fire extinguishment systems include the following major features:

a. Fire suppression water supplies, yard mains, hydrants, and valves
b. Automatic wet sprinklers
c. Water spray systems
d. Automatic preaction sprinklers
e. Manual preaction sprinklers
f. Automatic deluge system
g. Manual deluge systems
h. Automatic foam system

i. Carbon dioxide systems
j. Standpipes and hose reels
k. Portable extinguishers

The water supply for the fire protection system is taken from two freshwater storage tanks. Fire protection system pressure is normally maintained by a supply connection from the service water pump of the raw water system. If this interconnection is not able to maintain the system pressure, a motor-driven fire pump and two diesel engine-driven fire pumps automatically start in sequence to supply the fire protection systems.

Selected portions of the fire protection system are designed to meet seismic Category I criteria. These portions of the fire protection system provide manual fire fighting capability for the operation of any two hose stations in the plant areas containing equipment required for a safe shutdown. These protions of the fire protection system are normally supplied by the non-seismic Category I portion of the system. Two 100% capacity seismic Category 1 motor-driven fire pumps are provided to supply water to the seismic Category 1 system from two dedicated seismic Category I tanks in the event that the normal water supplies are not available.

The fire protection system is discussed further in Section 9.5.

1.2.5.5 Control Room HVAC System (Emergency Mode)

The control room HVAC system, when operating in the emergency mode, has been designed to maintain the control room envelope habitable during the postulated conditions resulting from a hypothetical LOCA, or any other occurrence generating high-level airborne radioactivity monitored at the control room outside air intakes.

The control room HVAC system, during the emergency mode, consists of two redundant trains whose major components include a supply air handling unit, return air fan, an emergency makeup air cleaning unit, and associated ductwork and accessories. The control room HVAC system design includes two remotely located outside air intakes. These intakes are connected to their respective makeup air cleaning unit, and are also cross-connected with the result that any emergency makeup air cleaning unit can use any of the two air intakes. Upon receipt of an engineered safety feature actuation signal-safety insection actuation signal (ESFAS-SIAS) and/or engineered safety feature actuation signal-control room emergency ventilation actuation signal (ESFAS-CREVAS), the control room normal HVAC system is automatically switched to the emergency mode, having dual intake provisions. The "cleaner" intake is manually selected by the operator. Switching from normal to the emergency mode can be accomplished manually from the control room.

Transfer to the emergency mode automatically diverts the normal minimum outside makeup air supply to the air cleaning unit and closes the redundant isolation dampers in the kitchen and toilet exhaust duct with subsequent tripping of kitchen and toilet exhaust fans. In addition, part of the recirculated air is also routed through the emergency air cleaning unit. The emergency makeup air cleaning unit is designed to remove radioiodine particulates and gases (elemental and methyl iodine) as the mixture of outside and recirculated air passes through a HEPA filter and carbon adsorber, thus maintaining the control room envelope iodine protection factor (IPF) at or above the allowable limit. Since there is no positive exhaust, the control room envelope exfiltrates to the outside and thus maintains the control room envelope at a positive pressure with respect to the surrounding areas. The cooling coil in the air handling unit in conjunction with the reheat coils maintain room temperature in the different areas of the envelope. The control room HVAC system is discussed further in Sections 6.4 and 9.4.

1.2.5.6 Containment Isolation System

The containment isolation system is provided to isolate the containment atmosphere from the outside environment in the event of a postulated accident that releases radioactive material inside the containment. The containment atmosphere is isolated from the outside environment by isolation valves and other barriers for all pipelines that penetrate the containment, unless such lines are required for service during the accident. The function of the containment isolation system is to provide an essentially leaktight barrier against the uncontrolled release of radioactivity to the environment and to limit the leakage to within the requirements of 10 CFR 100.

The containment isolation system is further discussed in Subsection 6.2.4.

1.2.5.7 Combustible Gas Control System

A hydrogen recombiner system is provided to control the concentration of hydrogen in the containment atmosphere to less than 4% following a postulated LOCA. Two independent, full-capacity intake and return piping systems are provided for each unit, to which one or both of the recombiners may be connected.

A post-LOCA purge system is also provided, however, it is not credited for accident mitigation. The post-LOCA purge system is valved into the recombiner piping system. The purge flow is passed through a demister, heating coil, high-efficiency prefilter, HEPA filter, a carbon adsorber, and a high-efficiency afterfilter before being released to the environment.

Additional information on combustible gas control is presented in Subsection 6.2.5. The post-LOCA purge system is discussed in Section 9.4.

1.2.5.8 Auxiliary Feedwater System

When the feedwater system is inoperable or unavailable, and the reactor coolant pressure and temperature are greater than 410 psia (28.8 kg/cm² A) and 350°F (177°C), respectively, the auxiliary feedwater system is used to supply water to the secondary side of the steam generators.

The system consists of two Class 1E motor-driven pumps and two diesel-driven pumps. During emergency plant cooldown, the auxiliary feedwater system can be used to supply feedwater to the steam generators for removal of decay and sensible heat from the RCS.

All pumps are started automatically by an auxiliary feedwater actuation signal (AFAS) actuated in case of a low water level in one steam generator or a signal from the diverse protection system (DPS).

The auxiliary feedwater system is further discussed in Subsection 10.4.9.

1.2.6 Instrumentation and Control

Automatic protection systems, control systems, and interlocks are provided, along with administrative controls, to ensure safe operation of the plant. Sufficient instrumentation and controls are supplied to provide manual operation as a normal backup control mode on all automatic systems.

The plant protection system (PPS) initiates a reactor trip if the reactor approaches prescribed safety limits, or provides an actuation signal to the engineered safety features systems when a fluid system or containment parameter approaches a prescribed limit.

Sufficient redundancy is installed to permit periodic testing of the PPS so that removal from service of any one protection system component or portion of the system will not preclude reactor trip, or other protective action when required. In addition, no single failure can preclude the PPS from providing a reactor trip or other protective action when required.

The protection system and associated instrumentation is separated from the control systems and their associated instrumentation such that failure or removal from service of any control system, component, or instrument channel will not inhibit the functioning of the protection system (see Chapter 7 for details).

1.2.6.1 Protection Systems

1.2.6.1.1 Reactor Protection System

The controllable reactor parameters are normally maintained within acceptable operating limits by the inherent characteristics of the reactor, the reactor regulating system (RRS), soluble boron concentration, and the plant operating procedures.

Four independent channels of the reactor protection system (RPS) normally monitor each of the selected plant parameters. The RPS logic is designed to initiate protective action whenever the signal of any two channels of a given parameter reach the preset limit. Should this occur, the power supplied to the control element drive mechanisms (CEDM) is interrupted, releasing the control element assemblies (CEA), which drop into the core to shut down the reactor. The two-out-of-four logic can be converted to two-out-of-three logic to allow one channel to be bypassed for testing maintenance or operation. The protection system is independent of and separate from the manual and automatic control systems except for a control element withdrawal prohibit (CWP).

1.2.6.1.2 Diverse Protection System

The diverse protection system (DPS) provides a simple yet diverse mechanism to increase the overall reliability of the plant protection system. The DPS augments reactor protection by utilizing a separate and diverse logic from the RPS for initiation of a reactor trip or for initiation of auxiliary feedwater. The DPS initiates a reactor trip when pressurizer pressure exceeds a predetermined value or initiates auxiliary feedwater to the steam generators when a steam generator water level drops below a predetermined level.

The DPS is provided with sensors and circuitry that are diverse from that of the RPS and the ESFAS. The DPS design uses a selective two-out-of-two logic to interrupt the power supplied to the CEDMs and thereby causes the CEAs to drop into the core. A selective two-out-of-two logic is also used by DPS to initiate auxiliary feedwater. The DPS is independent and separate from all other control systems.

1.2.6.2 Engineered Safety Features Actuation System

The NSSS engineered safety features actuation system (ESFAS) operates in a manner similar to the RPS to automatically actuate the engineered safety features (ESF) systems. Again, it has a selective two-out-of-four actuation logic that can be converted to a selective two-out-of-three logic. The NSSS ESFAS is completely independent of the control systems.

The balance-of-plant (BOP) ESFAS generates appropriate actuation signals which automatically actuate various process system equipment and component that perform BOP protective actions after receiving a signal from the BOP ESFAS or the operator.

The instrumentation and controls of the BOP ESFAS 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 BOP ESFAS from accomplishing their protective functions.

1.2.6.3 Systems Required for Safe Shutdown

The systems required for safe shutdown (SRSS) are those systems that must operate to shut down the reactor and maintain it in a safe shutdown condition. In many cases, the SRSS instrumentation and controls are used in the performance of normal plant operations, and as such cannot be exclusively identified for safe shutdown functions. The SRSS instruments and controls allow hot shutdown with or without offsite power.

These instrumentation and controls are designed so that their capability for achieving safe shutdown is not impaired during normal maintenance and testing.

Separation and independence of redundant instrumentation and controls provide assurance that a single failure within the system will not inhibit their safe shutdown functions.

1.2.6.4 Safety-Related Display Instrumentation

The safety-related display instrumentation provides information to the operator for performing the required safety functions. Monitored functions include the reactor coolant system, conditions in the containment and reactor, the reactor trip system, the ESF systems, the systems required for safe shutdown, and postaccident monitoring systems. These indications are available throughout planned operations, anticipated operational occurrences, and postaccident conditions.

1.2.6.5 Other Instrumentation and Control Systems

1.2.6.5.1 Reactor Control Systems

The reactor control systems are used for startup and shutdown of the reactor, and for adjustment of the reactor power in response to turbine load demand. The NSSS control systems are capable of following ramp load changes between 15% and 100% of full power at a rate of 5% per minute and a step change of 10% except as limited by xenon. This control is normally accomplished by automatic movement of CEAs in response to a change in reactor coolant temperature, with manual control capable of overriding the automatic signal at any time. If the reactor coolant temperature is different from a programmed value, the CEAs are adjusted until the difference is within the prescribed control band. Regulation of the reactor coolant temperature, in accordance with this program, maintains the main steam pressure within operating limits and matches reactor power to load demand.

The reactor is controlled by a combination of CEA motion and dissolved boric acid in the reactor coolant. Boric acid is used for reactivity changes associated with large but gradual changes in water temperature, xenon concentration, and fuel burnup. Addition of boric acid also provides an increased shutdown margin during the initial fuel loading and subsequent refuelings. The boric acid solution is prepared and stored at a temperature sufficient to prevent precipitation.

CEA movement provides changes in reactivity for shutdown or power changes. The CEAs are moved by CEDMs mounted on the reactor vessel head. The CEDMs are designed to permit rapid insertion of the CEAs into the reactor core by gravity. CEA motion can be initiated manually or automatically.

The pressure in the reactor coolant system is controlled by regulating the temperature of the coolant in the pressurizer where steam and water are held in thermal equilibrium. Steam is formed by the pressurizer heaters or condensed by the pressurizer spray to reduce variations caused by expansion and contraction of the reactor coolant due to system temperature changes.

Overpressure protection is provided by safety valves connected to the pressurizer and designed in accordance with the ASME Code, Section III. The discharge from the pressurizer safety valves is released underwater in the reactor drain tank, where it is cooled and condensed. Overpressure protection for the tank is provided by a rupture disc that relieves to the containment.

A steam bypass control system (SBCS) is used to dump steam in case of a large mismatch between the power being produced by the reactor and the power being used by the turbine. This allows the reactor to remain at power instead of tripping. Each steam generator's water level is maintained by a feedwater control system (FWCS). A reactor power cutback system (RPCS) is used to drop selected CEAs into the core to reduce reactor power rapidly during a large loss of load.

This allows the SBCS and FWCS to maintain the nuclear steam supply system in a stable condition, without a reactor trip, and without lifting any safety valves during loss-of-load transients.

1.2.6.5.2 Nuclear Instrumentation

The nuclear instrumentation includes excore and incore neutron flux detectors. Eight channels of excore instrumentation monitor the power. Two channels are provided for startup, two channels are provided for power control, and four channels are provided for the protection channels. The control channels are used to control the reactor power during power operations. The protection channels are used to provide inputs to the overpower, logarithmic power, departure from nucleate boiling ratio (DNBR), and local power density (LPD) trips in the RPS.

The incore instrumentation consists of self-powered detectors, distributed throughout the core, which provide information on flux distribution within the core.

1.2.6.5.3 Monitoring Systems

The plant monitoring system (PMS) performs general monitoring of the NSSS and balance of plant; logging, trending, and alarming of conditions are its major functions. The PMS does not perform any direct safety function. Part of the PMS is the core operating limit supervisory system (COLSS).

Temperature, pressure, flow, and liquid level are monitored to provide information on the plant operating condition to the operating personnel. Protection channels indicate the various parameters used for protective action as well as providing trip and pretrip alarms from the RPS.

The plant liquid and gaseous effluents are monitored to assure that they are maintained within applicable radioactivity limits. A complete description of the process and effluent radiation monitoring instrumentation is provided in Section 11.5.

1.2.7 Electrical Power Systems

1.2.7.1 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 Korea. There is no electrical system interconnection with any other utilities.

The main generator is an 1800 rpm, three-phase, 60 hertz, synchronous unit. The generator is connected through a coupling to the turbine shaft. 435

Power from the generator is stepped up from 22- to 345-kV by the main transformer and is supplied through a gas-insulated bus (SF6) to a 345kV switchyard common to both units. This switchyard is of gas-insulated construction and is connected to the offsite transmission network by four 345-kV transmission lines and two 345kV connections to the YGN 1&2 switchyard, which is located approximately 400 meters away.

1.2.7.2 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.

The power distribution system includes the Class 1E and non-Class 1E ac and dc power systems.

The Class 1E ac system for each unit consists of two independent and redundant divisions and four independent 120-volt vital ac instrumentation and control power supply systems. Each division includes 4.16-kV switchgear, 480-volt load centers, and motor control centers (MCCs). The vital ac instrumentation and control power supply systems include inverters, line voltage regulating transformers, and distribution panels.

The non-Class 1E ac system includes 13.8-kV switchgear, 4.16-kV switchgear, 480-volt load centers, motor control centers, and a non-Class 1E 120-volt vital ac power system.

The Class 1E dc system for each unit is supplied by four independent Class 1E 125-volt batteries and associated battery chargers. Non-Class 1E 250-volt and

125-volt batteries and associated battery chargers supply power for the non-Class 1E dc system loads.

These systems are discussed individually in Chapter 8.

1.2.8 Power Conversion System

The turbine-generator is an 1800 rpm tandem compound, six-flow, condensing, four-casing (one high-pressure casing and three low-pressure casings), 43-inch (109-cm) last-stage bucket, reheat unit with an electrohydraulic control including a digital control and monitoring (DCM) system. The turbine-generator's rated output is 1049 MW at 1.5 in. (38 mm) HgA backpressure.

Steam is supplied to the high-pressure turbine from two steam generators. The high-pressure turbine exhausts to the three low-pressure turbines through two moisture separator reheaters where moisture is removed and the steam is reheated in two stages. High-pressure extraction steam supplies the energy for the first stage of reheat, and main steam supplies the second stage reheat energy. The low-pressure turbines exhaust to three condensers where the steam is condensed.

Noncondensible gases are removed from the condenser by the steam jet air ejector and/or mechanical vacuum pumps and vented to the atmosphere. Heat from the steam is transferred to the circulating water in the condenser. The condenser also serves as a heat sink for the turbine bypass system during large load reduction. The circulating water system provides the condenser with a continuous supply of cooling water to remove the heat rejected from the turbine cycle. In the circulating water system, water from the Yellow Sea is pumped by vertical circulating water pumps through the condenser and returned to the Yellow Sea.

Condensed steam from the condenser is pumped by three condensate pumps through a condensate polishing system to maintain proper chemistry requirements and then heated through three stages of low-pressure closed feedwater heaters. There are three parallel strings of these heaters. The condensate then flows to a single, direct-contact, deaerating heater where it is further heated and deaerated, and drains by gravity to two deaerator storage tanks.

From the deaerator storage tanks, feedwater is pumped by two stages of feedwater pumps through the final three stages of closed feedwater heaters in two parallel strings and is then returned to the steam generators. Steam is extracted from the high- and low-pressure turbines and is supplied to the feedwater heaters. Drains from the high-pressure feedwater heaters and moisture separator/reheaters are cascaded to the next lower pressure heaters and/or to the deaerating heater for heating the feedwater. The low-pressure heater drains cascade to the next lower pressure heaters and/or to the condenser.

1.2.9 Fuel Handling and Storage Systems

The reactor is refueled by equipment that handles spent fuel underwater, from the time it is removed from the reactor vessel, until it is placed in the spent fuel storage racks or a cask for shipment from the site. Underwater transfer of spent fuel provides radiation shielding and reliable cooling for removal of decay heat.

The fuel handling operation is performed in two areas: the containment refueling pool, which is flooded for refueling, and the spent fuel pool and adjoining fuel transfer canal. The refueling pool and the fuel building transfer canal are connected by a fuel tranfer tube.

Spent fuel is removed from the reactor vessel and placed in the fuel transfer system by a refueling machine. It is then transferred from the reactor containment to the fuel building transfer canal through the fuel transfer tube. In the fuel transfer canal, spent fuel is removed from the fuel transfer system and is placed in spent fuel storage racks installed in the spent fuel pool. After a suitable decay period, spent fuel can be removed from the storage rack and loaded into a spent fuel shipping cask for removal to a reprocessing plant and/or ultimate storage.

New fuel is stored in vertical racks installed in the new fuel storage pit of the fuel building. Space is provided for storage of a minimum of one refueling batch. The criteria for the design of the new fuel storage pit and storage rack are discussed and evaluated in Subsection 9.1.1.

The stainless steel lined, reinforced-concrete spent fuel pool provides off-load storage for 20 years spent fuel plus one full core. Spent fuel 312
assemblies are stored in vertical racks designed to preclude criticality in nonborated cooling water. The designed criteria and evaluate of the fuel storage facility are discussed in Subsection 9.1.2. Control of the spent fuel pool water temperature during normal operation is accomplished by circulating the spent fuel pool water through heat exchangers cooled by the component cooling water system (CCWS). Purification and clarification of the spent fuel pool water are done by filters, strainers, and ion exchangers. A description and evaluation of the spent fuel pool cooling and cleanup 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 assemblies, and for the required assembly, disassembly, and storage of the reactor vessel head assembly and the reactor internals. This system includes a refueling machine located inside the containment above the refueling pool, the spent fuel transfer handling machine installed in the fuel building over the spent fuel pool and the fuel transfer canal, the fuel building overhead crane, the fuel transfer system carriage and upending mechanisms, the containment polar crane, the new fuel elevator, CEA elevator, the CEA change platform, the spent fuel handling tool, and various devices used for handling and storage of fuel assemblies (new and spent), the reactor vessel head, and the core internals.

1.2.10 Cooling Water and Other Auxiliary Systems

1.2.10.1 Shutdown Cooling System

The shutdown cooling system is used to reduce the temperature of the reactor coolant at a controlled rate from 350°F (176.7°C) to a refueling temperature of 125°F (51°C) and to maintain the proper reactor coolant temperature during refueling. This system utilizes the low-pressure safety injection pumps to circulate the reactor coolant through two shutdown heat exchangers and return it to the reactor coolant system through the low-pressure safety injection pumps. The component cooling water system supplies cooling water for the supplies cooling heat exchangers.

The shutdown cooling system is further discussed in Subsection 5.4.7.

1.2.10.2 Chemical and Volume Control System

The chemical and volume control system (CVCS) controls the purity, volume, and boric acid content of the reactor coolant.

The coolant purity level in the reactor coolant system is controlled by continuous purification of a bypass stream of reactor coolant. Water removed from the reactor coolant system is cooled in the regenerative heat exchanger. From there, the coolant flows to the letdown heat exchanger and then through a filter and an ion exchanger where corrosion and fission products are removed. It is then sprayed into the volume control tank and returned by the charging pumps to the regenerative heat exchanger where it is heated prior to return to the reactor coolant system.

The CVCS automatically adjusts the amount of reactor coolant in order to maintain a programmed level in the pressurizer. The level program partially compensates for changes in specific volume due to coolant temperature changes and reactor coolant pump controlled seal leakage (see Subsection 9.3.4.2 for details).

The CVCS controls the boric acid concentration in the coolant by a "feed and bleed" method where the purified letdown stream is diverted to a boron recovery section and either concentrated boric acid or demineralized water is sent to the charging pumps. The diverted coolant stream is processed by ion exchange and degasification and flows to a concentrator. The concentrator bottoms are sent to the refueling water tank for reuse as boric acid solution, and the distillate is first passed through an ion exchanger and then stored for reuse as demineralized water in the reactor makeup water tank. The refueling water tank is also used as a borated water source for the safety injection and containment spray systems.

1.2.10.3 Cooling Water Systems

Cooling water systems in operation at the facility include the essential service water system, component cooling water system, condensate storage facility, turbine building open and closed cooling water systems, circulating water system, chilled water systems, ultimate heat sink, and other plant water systems.

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 component cooling water system, essential chilled water system, condensate storage facility, and essential service water system are required for safe shutdown of the plant following a design-basis accident (DBA).

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

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

The condensate storage facility is designed to supply feedwater to the auxiliary feedwater system for filling the steam generators. The major components are two storage tanks, and associated piping and valves.

The ultimate heat sink is required for the dissipation of residual heat after reactor shutdown following an accident. The ultimate heat sink is the Yellow Sea, which guarantees 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 are described in Section 9.2.

1.2.10.4 Other Auxiliary Systems

The process auxiliary systems include the compressed air system, the process sampling systems, and the equipment and floor drainage systems. Process auxiliaries are discussed further in Section 9.3.

The non-safety-related 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 services. The compressed air system is required for normal plant operation but is not required for safe shutdown.

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 treatment systems. Monitoring capability is provided to ensure that inadvertent releases of radioactivity are prevented.

1.2.10.5 Ventilation Systems

Ventilation systems have been provided for normal plant operation and for DBAs. The auxiliary building, turbine building, fuel building, electrical equipment rooms, diesel generator rooms, radwaste building, intake structures/pump house, control room, access control building, alternate AC diesel generator building, and the containment are provided with HVAC systems. These ventilation systems have been designed to provide suitable environments for equipment and personnel. Where appropriate, ventilation zones have been arranged within buildings to allow the flow of air from clean areas to areas of potentially greater contamination prior to final exhaust. These systems are described further in Sections 6.4 and 9.4.

1.2.11 Radwaste Systems

The radwaste systems are designed to safely control potentially radioactive liquid, gaseous, and solid wastes. The systems consist of three principal systems:

a. Liquid radwaste system

b. Gaseous radwaste system

c. Solid radwaste system

The design of the radwaste systems ensures that the total offsite dose resulting from radioactive releases is as low as is reasonably achievable.

1.2.11.1 Liquid Radwaste System

The liquid radwaste system (LRS) is designed to purify radioactive and chemical liquid wastes for maximum recycle. The system collects and segregates the waste on the basis of dissolved and suspended solids for optimal treatment. For wastes low in total solids, the treatment method is particulate removal by filters followed by dissolved impurities removal by ion exchangers. For wastes high in total solids, the treatment method is particulate removal by filters followed by concentration in an evaporator. Evaporator distillate is routed through ion exchangers with other low-solids waste. The ion exchangers effluent is recycled as reactor makeup water or condensate makeup or discharged to the sea after appropriate monitoring. See Section 11.2 for a detailed description of the LRS.

1.2.11.2 Gaseous Radwaste System

The gaseous radwaste system (GRS) is designed to treat the high-activity, hydrogen-rich gases from the reactor coolant. The gases are collected in a header, dried, and adsorbed on charcoal delay beds to permit radioactive decay. After passing through the charcoal beds, the gases are filtered, monitored, and discharged to the plant ventilation system for dilution and release to the environment. See Section 11.3 for a detailed description of the GRS.

1.2.11.3 Solid Radwaste System

The solid radwaste system (SRS) is designed to process radioactive solids, sludges, and concentrated liquid waste for offsite shipment and disposal. These wastes include spent resins, tank bottom sludge, CVCS boric acid concentrator and LRS evaporator concentrates, spent filter cartridges, and 1
other miscellaneous contaminated solid refuse. Concentrates are dried to powder for volume reduction. Concentrates powder and other wet wastes are mixed with a suitable solidification agent and packaged in a container for disposal. Miscellaneous low-activity solids are compacted into drums for disposal. A detailed system description is provided in Section 11.4.

KOREA ELECTRIC POWER CORPORATION
YONGGWANG 3 & 4
FSAR

SITE DEVELOPMENT

Figure 1.2-1

KOREA ELECTRIC POWER CORPORATION
YONGGWANG 3 & 4
FSAR

PROPERTY DEVELOPMENT

Figure 1.2-2

KOREA HYDRO & NUCLEAR POWER COMPANY
YONGGWANG 3 & 4
FSAR

PLANT ARRANGEMENT ROOF PLAN

Figure 1.2-3

KOREA HYDRO & NUCLEAR POWER COMPANY
YONGGWANG 3 & 4
FSAR

GENERAL ARRANGEMENT
SITE PLOT PLAN

Figure 1.2-4

KOREA HYDRO & NUCLEAR POWER COMPANY
YGN 3 & 4 FSAR

GENERAL ARRANGEMENT
TURBINE. BLDG. BSMT. EL. 73'-0"
Figure 1.2-5

KOREA HYDRO & NUCLEAR POWER COMPANY
YGN 3 & 4 FSAR

GENERAL ARRANGEMENT
TURBINE BLEG. GRAND. EL. 100'-6"

Figure 1.2-6

KOREA HYDRO & NUCLEAR POWER COMPANY
YONGGWANG 3 & 4
FSAR

GENERAL ARRANGEMENT
TURBINE BLDG. OPER. EL. 135'-0"

Figure 1.2-7

KOREA HYDRO & NUCLEAR POWER COMPANY
YONGGWANG 3 & 4
FSAR

GENERAL ARRANGEMENT
TURBINE BLDG. DEAER. EL. 164'-0"

Figure 1.2-8

KOREA HYDRO & NUCLEAR POWER COMPANY
YONGGWANG 3 & 4
FSAR

GENERAL ARRANGEMENT
TURBINE BLDG. SECT. "A-A"

Figure 1.2-9

KOREA HYDRO & NUCLEAR POWER COMPANY
YONGGWANG 3 & 4
FSAR

GENERAL ARRANGEMENT
TURBINE BLDG. SECT. "B-B"

Figure 1.2-10

KOREA HYDRO & NUCLEAR POWER COMPANY
YONGGWANG 3 & 4
FSAR

GENERAL ARRANGEMENT
TURBINE BLDG. SECT. "C-C"

Figure 1.2-11

KOREA HYDRO & NUCLEAR POWER COMPANY
YONGGWANG 3 & 4
FSAR

GENERAL ARRANGEMENT
TURBINE BLDG. SECT. "D-D"

Figure 1.2-12

KOREA ELECTRIC POWER CORPORATION
YONGGWANG 3 & 4
FSAR

GENERAL ARRANGEMENT
TURBINE BLDG. SECT. "E-E"

Figure 1.2-13

KOREA ELECTRIC POWER CORPORATION
YONGGWANG 3 & 4
FSAR

GENERAL ARRANGEMENT
PRI. AUX. BLDG. EL. 58'-0"

Figure 1.2-14

KOREA HYDRO & NUCLEAR POWER COMPANY
YGN 3 & 4 FSAR

GENERAL ARRANGEMENT
PRI. AUX. BLDG. EL. 77'-0"

Figure 1.2-15

KOREA HYDRO & NUCLEAR POWER COMPANY
YGN 3 & 4 FSAR

GENERAL ARRANGEMENT
PRI. AUX. BLDG. GRND. EL. 100'-6"

Figure 1.2-16

KOREA ELECTRIC POWER CORPORATION
YONGGWANG 3 & 4
FSAR

GENERAL ARRANGEMENT
PRI. AUX. BLDG. EL. 125'-0"

Figure 1.2-17

KOREA HYDRO & NUCLEAR POWER COMPANY
YGN 3 & 4 ESAR

GENERAL ARRANGEMENT
PRI. AUX. BLDG. EL. 144'-0"

Figure 1.2-18

KOREA ELECTRIC POWER CORPORATION
YONGGWANG 3 & 4
FSAR

GENERAL ARRANGEMENT : PRI.
AUX. BLDG. EL. 165'-0"

Figure 1.2-19

KOREA HYDRO & NUCLEAR POWER COMPANY
YONGGWANG 3 & 4 FSAR

GENERAL ARRANGEMENT
PRI. AUX. BLDG. EL.182'-0"

Figure 1.2-20

KOREA ELECTRIC POWER CORPORATION
YONGGWANG 3 & 4
FSAR

GENERAL ARRANGEMENT
PRI. AUX. BLDG. SECT. "A-A"

Figure 1.2-21

KOREA ELECTRIC POWER CORPORATION
YONGGWANG 3 & 4
FSAR

GENERAL ARRANGEMENT
PRI. AUX. BLDG. SECT. "B-B"

Figure 1.2-22

KOREA HYDRO & NUCLEAR POWER COMPANY
YONGGWANG 3 & 4 FSAR

GENERAL ARRANGEMENT
PRI. AUX. BLDG. SECT. "C-C"

Figure 1.2-23

KOREA ELECTRIC POWER CORPORATION
YONGGWANG 3 & 4
FSAR

GENERAL ARRANGEMENT
PRI. AUX. BLDG. SECT. "D-D"

Figure 1.2-24

KOREA HYDRO & NUCLEAR POWER COMPANY
YGN 3 & 4 FSAR

GENERAL ARRANGEMENT
PRI. AUX. BLDG, SECT. "E-E"

Figure 1.2-25

KOREA ELECTRIC POWER CORPORATION
YONGGWANG 3 & 4
FSAR

GENERAL ARRANGEMENT
SEC. AUX. BLDG. EL. 58'-0" & 77'-0"

Figure 1.2-26

KOREA HYDRO & NUCLEAR POWER COMPANY
YGN 3 & 4
FSAR

GENERAL ARRANGEMENT
SEC. AUX. BLDG. EL. 100'-6" & 125'-0"

Figure 1.2-27

KOREA ELECTRIC POWER CORPORATION
YONGGWANG 3 & 4
FSAR

GENERAL ARRANGEMENT
SEC. AUX. BLDG. EL. 144'-0"

Figure 1.2-28

KOREA ELECTRIC POWER CORPORATION
YONGGWANG 3 & 4
FSAR

GENERAL ARRANGEMENT
SEC. AUX. BLDG. SECT. "A-A"

Figure 1.2-29

KOREA HYDRO & NUCLEAR POWER COMPANY
YGN 3 & 4
FSAR

GENERAL ARRANGEMENT
SEC. AUX. BLDG SECT. "B-B"

Figure 1.2-30

KOREA HYDRO & NUCLEAR POWER COMPANY
YGN 3 & 4
FSAR

GENERAL ARRANGEMENT
SEC. AUX. BLDG SECT. "C-C" & "D-D"

Figure 1.2-31

KOREA ELECTRIC POWER CORPORATION
YONGGWANG 3 & 4
FSAR

GENERAL ARRANGEMENT
CONTAINMENT BLDG. EL. 86'-0"

Figure 1.2-32

KOREA ELECTRIC POWER CORPORATION
YONGGWANG 3 & 4
FSAR

GENERAL ARRANGEMENT
CONTAINMENT BLDG. EL. 100'-0"

Figure 1.2-33

KOREA ELECTRIC POWER CORPORATION
YONGGWANG 3 & 4
FSAR

GENERAL ARRANGEMENT
CONTAINMENT BLDG. EL. 122'-0"

Figure 1.2-34

KOREA ELECTRIC POWER CORPORATION
YONGGWANG 3 & 4
FSAR

GENERAL ARRANGEMENT
CONTAINMENT BLDG. EL. 142'-0"

Figure 1.2-35

KOREA ELECTRIC POWER CORPORATION
YONGGWANG 3 & 4
FSAR

GENERAL ARRANGEMENT
CONTAINMENT BLDG. SECT. "A-A"

Figure 1.2-36

KOREA ELECTRIC POWER CORPORATION
YONGGWANG 3 & 4
FSAR

GENERAL ARRANGEMENT
CONTAINMENT BLDG. SECT. "B-B"

Figure 1.2-37

KOREA ELECTRIC POWER CORPORATION
YONGGWANG 3 & 4
FSAR

GENERAL ARRANGEMENT
FUEL BLDG.

Figure 1.2-38

KOREA ELECTRIC POWER CORPORATION
YONGGWANG 3 & 4
FSAR

GENERAL ARRANGEMENT
FUEL BLDG, TUNNELS & WALKWAY

Figure 1.2-39

KOREA HYDRO & NUCLEAR POWER COMPANY
YGN 3 & 4
FSAR

GENERAL ARRANGEMENT : ACCESS CONTROL
BULD. EL. 77'-0" & 100'-6"

Figure 1.2-40

KOREA HYDRO & NUCLEAR POWER COMPANY
YGN 3 & 4
FSAR

GENERAL ARRANGEMENT : ACCESS CONTROL
BULD. EL. 120'-0" & 135'-0"

Figure 1.2-41

KOREA ELECTRIC POWER CORPORATION
YONGGWANG 3 & 4
FSAR

GENERAL ARRANGEMENT
ACCESS CNTRL. BLDG.
EL. 150'-0" & 165'-0" (UNIT 3)

Figure 1.2-42

KOREA ELECTRIC POWER CORPORATION
YONGGWANG 3 & 4
FSAR

GENERAL ARRANGEMENT
ACCESS CNTRL. BLDG.
EL. 150'-0" & 165'-0" (UNIT 4)

Figure 1.2-43

KOREA ELECTRIC POWER CORPORATION
YONGGWANG 3 & 4
FSAR

GENERAL ARRANGEMENT
RADWASTE BLDG. EL. 100'-6" & 113'-0"

Figure 1.2-44

KOREA ELECTRIC POWER CORPORATION
YONGGWANG 3 & 4
FSAR

GENERAL ARRANGEMENT : RADWASTE BLDG. EL. 123'-0" & 135'-0"

Figure 1.2-45

KOREA HYDRO & NUCLEAR POWER COMPANY
YGN 3 & 4
FSAR

GENERAL ARRANGEMENT : RADWASTE
BLDG. EL. 150'-0" & 170'-0"

Figure 1.2-46

KOREA ELECTRIC POWER CORPORATION
YONGGWANG 3 & 4
FSAR

GENERAL ARRANGEMENT : RADWASTE BLDG.

Figure 1.2-47

KOREA ELECTRIC POWER CORPORATION
YONGGWANG 3 & 4
FSAR

GENERAL ARRANGEMENT
RADWASTE BLDG. SECTIONS

Figure 1.2-48

KOREA ELECTRIC POWER CORPORATION
YONGGWANG 3 & 4
FSAR

GENERAL ARRANGEMENT
INTAKE STRUCTURE / PUMP HOUSE
EL. 30'-3"

Figure 1.2-49

KOREA HYDRO & NUCLEAR POWER COMPANY
YGN 3 & 4 FSAR

GENERAL ARRANGEMENT
INTAKE STRUCTURE / PUMP HOUSE
EL. 80'- 0"

Figure 1.2 - 50

KOREA ELECTRIC POWER CORPORATION
YONGGWANG 3 & 4
FSAR

GENERAL ARRANGEMENT
INTAKE STRUCTURE SECT. "A-A"

Figure 1.2-51

KOREA HYDRO & NUCLEAR POWER COMPANY
YONGGWANG 3 & 4
FSAR

GENERAL ARRANGEMENT
INTAKE STRUCTURE
SECTION "B-B"

Figure 1.2-52

KOREA ELECTRIC POWER CORPORATION
YONGGWANG 3 & 4
FSAR

GENERAL ARRANGEMENT
ESW INTAKE STRUCTURE

Figure 1.2-53

KOREA ELECTRIC POWER CORPORATION
YONGGWANG 3 & 4
FSAR

GENERAL ARRANGEMENT
HEAT EXCHANGER BLDG.

Figure 1.2-54

KOREA ELECTRIC POWER CORPORATION
YONGGWANG 3 & 4
FSAR

GENERAL ARRANGEMENT
AAC DIESEL GENERATOR BUILDING
(Sheet 1 of 2)

Figure 1.2-55

KOREA ELECTRIC POWER CORPORATION
YONGGWANG 3 & 4
FSAR

GENERAL ARRANGEMENT
AAC DIESEL GENERATOR BUILDING
(Sheet 2 of 2)

Figure 1.2-55

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 those of 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 presents a summary of the YGN 3&4 reactor core and coolant system characteristics and a similar summary of Arkansas Nuclear One (ANO) Unit 2 and Palo Verde Nuclear Generation Station (PVNGS) Unit 2.

The ANO 2 and PVNGS 2 designs were selected for comparison in Table 1.3-1 because of the basic similarity of the reactor core and coolant systems.

In Table 1.3-2, YGN 1&2, Byron/Braidwood, and PVNGS 2 have been listed for comparison of plant features other than the NSSS.

1.3.2 Comparison of Final and Preliminary Information

Table 1.3-3 contains a discussion of significant changes that have been made in plant design since submittal of the YGN 3&4 PSAR and Amendments 1, 2, and 3.

TABLE 1.3-1 (Sh. 1 of 11)

COMPARISON OF REACTOR CORE AND
COOLANT SYSTEM PARAMETERS WITH SIMILAR FACILITY DESIGNS

ITEM	REFERENCE SECTION	PVNGS 2	ANO 2	YGN 3&4 (GUARDIAN/PLUS7)
Hydraulic and Thermal Design Parameters				
Rated core heat output (MWt)	4.4	3,800	2,815	2,815
Rated core heat output (Btu/hr)	4.4	$12,970 \times 10^6$	$9,608 \times 10^6$	$9,608 \times 10^6$
Heat generated in fuel (%)	4.4	97.5	97.5	97.5
System pressure, nominal (psia)	4.4	2,250	2,250	2,250
System pressure, minimum steady state (psia)	4.4	2,200	2,200	2,200
Hot channel factors				
Heat flux (F_q)	4.4	2.35	2.35	2.35/2.42
Enthalpy rise (F_H) (outlet enthalpy = 699)	4.4	1.56	1.55	1.55/1.60
DNB ratio at nominal conditions	4.4	1.79 (CE-1)	2.14 (W-3)	2.07 (CE-1)/2.27(KCE-1) 563
Coolant flow(based on nominal RCS flow rate)				
Total flow rate (lb/hr)	4.4	164.0×10^6	120.4×10^6	121.5×10^6
Effective flow rate for heat transfer (lb/hr)	4.4	157.4×10^6	116.2×10^6	117.9×10^6
Effective flow area for heat transfer (ft^2)	4.4	60.9	44.7	44.8/46.2
Average velocity along fuel rods (ft/sec)	4.4	16.4	16.4	16.7/16.2

TABLE 1.3-1 (Sh. 2 of 11)

ITEM	REFERENCE SECTION	PVNGS 2	ANO 2	YGN 3&4 (GUARDIAN/PLUS7)
Average mass velocity ($lb/h\text{-}ft^2$)	4.4	$2.58x10^6$	$2.60x10^6$	$2.63x10^6/2.55x10^6$
Coolant temperatures (°F)(based on nominal RCS flow rate)				
Nominal inlet	4.4	564.5	553	564.5
Design inlet	4.4	568	556.5	567.5
Average rise in vessel	4.4	56	58.5	56.5
Average rise in core	4.4	59	60.5	58
Average in core	4.4	594	583.75	594
Average in vessel	4.4	593	582.75	593
Nominal outlet of hot channel	4.4	653	652.6	646/651
Average film coefficient ($Btu/h\text{-}ft^2\text{-}°F$)	4.4	6,300	6,170	6,300/6,060
Average film temperature difference (°F)	4.4	30	31	28.5/30.3
Heat transfer at 100 % rated power				
Active heat transfer surface area (ft^2)	4.4	68,600	51,000	52,100/51,023
Average heat flux ($Btu/h\text{-}ft^2$)	4.4	184,400	185,000	179,750/183,545
Maximum heat flux ($Btu/h\text{-}ft^2$)	4.4	433,000	433,800	422,000/444,648
Average thermal output (kW/ft)	4.4	5.40	5.41	5.26
Maximum thermal output (kW/ft)	4.4	12.7	12.7	12.4/12.7

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TABLE 1.3-1 (Sh. 3 of 11)

ITEM	REFERENCE SECTION	PVNGS 2	ANO 2	YGN 3&4 (GUARDIAN/PLUS7)
Maximum clad surface temperature at nominal pressure (°F)	4.4	656	657	656.7
Fuel center temperature (°F)	4.4	3,290	3,420	3,131/3,031
Core Mechanical Design Parameters				
Fuel assemblies				
Design	4.2	CEA	CEA	CEA
Rod pitch (in.)	4.2	0.506	0.506	0.506
Cross section dimensions (in.)	4.2	7.972x7.972	7.972x7.972	7.972x7.972 / 7.964x7.964
Fuel weight as UO_2 (lb)	4.2	$257.1x10^3$	$183.6x10^3$	$189.9x10^3$ / $190.7x10^3$
Total weight (lb)	4.2	346,076	256,827	254,388 / 249,499
Number of grids per assembly	4.2	11	12	11/12
Fuel rods				
Number	4.2	56,876	40,644	41,772
Outside diameter (in.)	4.2	0.382	0.382	0.382/0.374
Diametral gap (in.)	4.2	0.007	0.007	0.007/0.0065
Clad thickness (in.)	4.2	0.025	0.025	0.025/0.0225
Clad material	4.2	Zircaloy-4	Zircaloy-4	ZIRLO/ZIRLO or M5

TABLE 1.3-1 (Sh. 4 of 11)

ITEM	REFERENCE SECTION	PVNGS 2	ANO 2	YGN 3&4 (GUARDIAN/PLUS7)
Fuel pellets				
Material	4.2	UO_2 sintered	UO_2 sintered	UO_2 sintered
Diameter (in.)	4.2	0.325	0.325	0.325/0.3225
Length (in.)	4.2	0.390	0.390	0.390/0.387
Control assemblies				
Neutron absorber	4.2	B_4C	B4C/Ag-In-Cd	B_4C/Inconel
Cladding material	4.2	Inconel 625	NiCrFe alloy	Inconel 625
Clad thickness	4.2	0.035	0.035	0.035
Number of assemblies, full/part-length	4.2	76/13	73/8	65/8(part strength)
Number of rods per assembly	4.2	4 or 12 (4 part-length rods per part-length assembly)	5	4 or 12 (4 part-strength rods per part-strength assembly)
Nuclear Design Data				
Structural characteristics				
Core diameter, equivalent (in.)	4.2	143.6	123	123
Core height, active fuel (in.)	4.2	150	150	150
H/U, unlimited assembly (hot)	4.3	4.20	4.19	2.12 (H_2O/UO_2 volume ratio)
Number of fuel assemblies	4.2	241	177	177

TABLE 1.3-1 (Sh. 5 of 11)

ITEM	REFERENCE SECTION	PVNGS 2	ANO 2	YGN 3&4 (GUARDIAN/PLUS7)
UO_2 and poison rods per assembly	4.2	236	236	236
Performance characteristics				
Loading technique	4.3	5-region mixed central zone	3-batch mixed central zone	3-batch mixed central zone
Fuel enrichment (wt%)				
Region 1	4.2	1.92	1.93	4.00 and 4.50
Region 2	4.2	1.92 and 2.78	2.27	4.00 and 4.50
Region 3	4.2	1.92 and 2.78	2.94	4.00 and 4.50
Region 4	4.2	2.78 and 3.30	N/A	N/A
Region 5	4.2	2.78 and 3.30	N/A	N/A
Control characteristics effective multiplication (beginning of life)				
Cold, no power, clean	4.3	1.193	1.182	1.220/1.211
Hot, no power, clean	4.3	1.133	1.136	1.158/1.153
Hot, full power, Xe equilibrium	4.3	1.078	1.075	1.099/1.095
Control assemblies				
Total rod worth (hot), (%$\triangle\rho$)	4.3	16.76	12.3	15.39/15.94

TABLE 1.3-1 (Sh. 6 of 11)

ITEM	REFERENCE SECTION	PVNGS 2	ANO 2	YGN 3&4 (GUARDIAN/PLUS7)
Boron concentrations for criticality:				
Zero power no rods inserted, clean (ppm) Cold/Hot	4.3	1,084/1,018	1,004/987	1,640/1,816 /1,536/1,713
At power with no rods inserted, clean/ equilibrium Xe (ppm)		911/657	870/612	1,625/1,271 /1,535/1,186
Kinetic characteristics, range over life				
Moderator temperature coefficient ($\triangle\rho$/°F)	4.3	-0.7×10^{-4} to -2.5×10^{-4}	-0.5×10^{-4} to -3.1×10^{-4}	-0.81×10^{-4} to -3.47×10^{-4} / -0.77×10^{-4} to -3.43×10^{-4}
Moderator pressure coefficient ($\triangle\rho$/psi)	4.3	$+0.64 \times 10^{-6}$	$+0.45 \times 0^{-6}$ to $+2.97 \times 10^{-6}$	$+0.11 \times 10^{-6}$
Moderator void coefficient ($\triangle\rho$/% void)	4.3	-0.24×10^{-3}	-0.28×10^{-3} to 1.47×10^{-3}	-0.47×10^{-3}
Doppler coefficient ($\triangle\rho$/ °F)	4.3	-1.18×10^{-5} to -1.66×10^{-5}	-1.18×10^{-5} to -1.78×10^{-5}	-1.60×10^{-5} to -1.76×10^{-5} / -1.60×10^{-5} to -1.77×10^{-5}
Reactor Coolant System - Code Requirements				
Component				
Reactor vessel	5.2	ASME III, Class 1	ASME III Class A	ASME III Class 1

TABLE 1.3-1 (Sh. 7 of 11)

ITEM	REFERENCE SECTION	PVNGS 2	ANO 2	YGN 3&4
Steam generator				
Tube side	5.2	ASME III, Class 1	ASME III, Class A	ASME III, Class 1
Shell side	5.2	ASME III, Class 2	ASME III, Class A	ASME III, Class 2
Pressurizer	5.2	ASME III, Class 1	ASME III, Class 1	ASME III, Class 1
Pressurizer safety valves	5.2	ASME III, Class 1	ASME III, Class 1	ASME III, Class 1
Reactor coolant piping	5.2	ASME III, Class 1	ASME III, Class 1	ASME III, Class 1
Principal Design Parameters of the Reactor Coolant System				
Operating pressure (psia)	5.1	2,250	2,250	2,250
Reactor inlet temperature (°F)	5.1	564.5	553.5	564.5
Reactor outlet temperature (°F)	5.1	621.2	612.5	621.2
Number of loops	5.1	2	2	2
Design pressure (psia)	5.1	2,500	2,500	2,500
Design temperature (°F)	5.1	650	650	650
Total coolant volume (ft^3) (without pressurizer)	5.1	12,353	9,376	10,800

TABLE 1.3-1 (Sh. 8 of 11)

ITEM	REFERENCE SECTION	PVNGS 2	ANO 2	YGN 3&4
Principal Design Parameters of the Reactor Vessel				
Material	5.2	SA-533, Grade B, Class 1 (shell); clad with austenitic SS	SA-533, Grade B, Class 1 low alloy steel; clad with Type 304 austenitic SS	SA-508, Class 1, 2, and 3 (shell); clad with austenitic SS
Design pressure (psia)	5.4	2,500	2,500	2,500
Design temperature (°F)	5.1	650	650	650
Operating pressure (psia)	5.3	2,250	2,250	2,250
Inside diameter of shell (in.)	5.3	182-1/4	157	162
Outside diameter across nozzles (in.)	5.3	267-1/4	238	263-5/8
Overall height of vessel and enclosure head to top of CEDM nozzle (ft-in.)	5.3	48-0	43-4-1/6	46-11-1/8
Minimum clad thickness (in.)	5.3	1/8	1/8	1/8
Principal Design Parameters of the Steam Generators				
Number of Units	5.4	2	2	2
Type	5.4	VerticalU-tube with integral moisture sepa-rator	Vertical U-tube with integral moisture sepa-rator	Vertical U-tube with integral moisture sepa-rator

TABLE 1.3-1 (Sh. 9 of 11)

ITEM	REFERENCE SECTION	PVNGS 2	ANO 2	YGN 3&4
Tube material	5.2	NiCrFe alloy	NiCrFe alloy	NiCrFe alloy
Shell material	5.2	SA-533 Gr. B Class I and SA-516 Gr. 70	SA-533 Gr. B Class I and SA-516 Gr. 70	SA-533 Gr. B Class I and SA-516 Gr. 70
Tube side design pressure (psia)	5.4	2,500	2,500	2,500
Tube side design temperature (°F)	5.4	650	650	650
Tube side design flow each (lb/h)	5.4	82.0×10^6	60.2×10^6	60.75×10^6
Shell side design pressure (psia)	5.4	1,270	1,100	1,270
Shell side design temperature (°F)	5.4	575	560	575
Operating pressure, tube side, nominal (psia)	5.4	2,250	2,250	2,250
Operating pressure, shell side, maximum (psia)	5.4	1,070	1,000	1,070
Maximum moisture at outlet at full load (%)	5.4	0.25	0.2	0.25
Steam pressure at full power (psia)	4.4	1,070	900	1,070
Steam temperature at full power (°F)	4.4	552.9	531.95	552.9
Principal Design Parameters of the Reactor Coolant Pumps				
Number of units	5.4	4	4	4

TABLE 1.3-1 (Sh. 10 of 11)

ITEM	REFERENCE SECTION	PVNGS 2	ANO 2	YGN 3&4
Type	5.4	Vertical, single-stage centrifugal with bottom suction and horizontal discharge	Vertical, single-stage centrifugal with bottom suction and horizontal discharge	Vertical, single-stage centrifugal with bottom suction and horizontal discharge
Design pressure (psia)	5.4	2,500	2,500	2,500
Design temperature (°F)	5.4	650	650	650
Operating pressure, nominal (psia)	5.4	2,250	2,250	2,250
Suction temperature (°F)	5.4	564.5	535.5	564.5
Design capacity (gal/min)	5.4	111,400	80,000	82,500
Design head (ft)	5.4	363	275	340
Motor type	5.4	ac induction, single speed	ac induction, single speed	ac induction, single speed
Motor rating (hp)	5.4	12,000	6,500	8,800
<u>Principal Design Parameters of the Reactor Coolant Piping</u>				
Material	5.2	SA-516, Gr. 70 with SS clad	SA-516, Gr. 70 with SS clad	SA-516, Gr. 70 with SS clad
Hot leg OD (in.)	5.4	42	42	42

TABLE 1.3-1 (Sh. 11 of 11)

ITEM	REFERENCE SECTION	PVNGS 2	ANO 2	YGN 3&4
Cold leg OD (in.)	5.4	30	30	30
Piping between pump and steam generator ID (in.)	5.4	30	30	30

TABLE 1.3-2 (Sh. 1 of 5)

COMPARISON OF PLANT COMPONENTS OTHER THAN NSSS WITH SIMILAR FACILITY DESIGNS

SYSTEM/PARAMETER	REFERENCE SECTION	YGN 1&2	BYRON/ BRAIDWOOD	PVNGS 2	YGN 3&4
Containment type	3.8	Steel-lined post-tensioned prestressed concrete cylinder, hemispherical dome roof	Steel-lined post-tensioned prestressed concrete cylinder, shallow dome roof dome	Steel-lined post-tensioned prestressed concrete cylinder, curved dome	Steel-lined post-tensioned prestressed concrete cylinder, hemispherical dome roof
Leak rate (%/day)		0.1 (24 hr) 0.05 (after 24 hr)	0.1 (24 hr)	0.1 (24 hr)	0.1 (24 hr) 0.05 (after 24 hr)
Design pressure (psig)		60	50	60	54
Free volume (10^6 ft^3)		2.15	2.75	2.6	2.727
Diameter/height (ft)		130/195	140/222	146/206	144/216
Containment spray	6.2, 6.5				
Number of pumps		2	2	2	2
Design capacity, each (gal/min)		3,000 (injection) 3,650 (recirculation)	3,285 (injection) 3,795 (recirculation)	3,890 (inj.)	3,500 4,400 (recirculation)

TABLE 1.3-2 (Sh. 2 of 5)

SYSTEM/PARAMETER	REFERENCE SECTION	YGN 1&2	BYRON/ BRAIDWOOD	PVNGS 2	YGN 3&4
Spray additive		NaOH	NaOH	Hydrazine	Hydrazine
Containment coolers (safety related)	6.2.2.2				
Type		Fan coolers	Fan coolers	N/A	Fan coolers
Number of units		4	4		4
Capacity (10^6 Btu/hr)/Unit		50	132		151
Onsite power systems, ac	8.3.1				
Generator prime mover		Diesel engine	Diesel engine	Diesel engine	Diesel engine
Number of units		2	2	2	2
Capacity, each (kW)		7,000	5,500	5,720	6,500
Ultimate heat sink	9.2.5				
Type		Yellow Sea	Cooling tower/ pond	Spray ponds	Yellow Sea
Backup		None	Onsite deep wells/aux. cooling panel	Domestic water system, makeup water reservoir	None

TABLE 1.3-2 (Sh. 3 of 5)

SYSTEM/PARAMETER	REFERENCE SECTION	YGN 1&2	BYRON/ BRAIDWOOD	PVNGS 2	YGN 3&4
Condensate storage facility	9.2.6				
Capacity (10^3 gal) (for each unit)		900	500	550	1000
Plant fire protection	9.5.1				
Water source		Tank storage	Cooling tower/ lake	Tank storage	Fresh water storage tank
Backup source		Condensate storage tank	Essential service water system	None	Seismic Category 1 fire water tanks
Emergency diesel generators	8.3, 9.5.4				
Fuel oil storage capacity per diesel operating at full power (days)		7	7	7	7
Alternate AC diesel generator	8.3, 9.5.12				
Fuel oil storage capacity per diesel operating at full power (days)		None	None	None	7

TABLE 1.3-2 (Sh. 4 of 5)

SYSTEM/PARAMETER	REFERENCE SECTION	YGN 1&2	BYRON/ BRAIDWOOD	PVNGS 2	YGN 3&4
Turbine-generator	10.2				
Output (MWe)		996.8	1,175	1,304	1,049
Main steam supply	10.3				
Total steam flow (10^6 lb/hr)		12.29	15.13	17.5	12.72
Steam generator outlet pressure (psia)		964	990	1,070	1,070
Steam generator outlet temp. (°F)		540	558.1	552.9	552.9
Main condensers	10.4.1				
Type		Single pressure	Single pressure	Multisection multipressure	Single pressure
Pressure (in. HgA)		1.5	3.5	4.5	1.5
Turbine bypass Capacity (% of rated load main steam flow)	10.4.4	40 (to condenser) 30 (to atmosphere)	40 (to condenser)	55	40 (to condenser) 15 (to atmosphere)

TABLE 1.3-2 (Sh. 5 of 5)

SYSTEM/PARAMETER	REFERENCE SECTION	YGN 1&2	BYRON/ BRAIDWOOD	PVNGS 2	YGN 3&4
Circulating water	10.4.5				
Type		Once-through, Yellow Sea	Cooling tower/ cooling pond	Cooling tower	Once-through, Yellow Sea
Auxiliary feedwater	10.4.9				
Pump prime movers		1 steam turbine 2 electric motors	1 electric motor 1 diesel-driven	1 steam turbine 2 electric motors	2 diesel-driven 2 electric motors
Rated flow rate, each (gal/min)		935 (turbine) 450 (motor)	890 (motor) 840 (diesel)	750 (S.C.I.) 710 (N.S.C.I.)	300 (each)
Radwaste systems					
Liquid radwaste system tankage	11.2	190,000 gal (2 units)	300,000 gal (2 units)	335,000 gal (2 units)	575,000 gal (2 units)
Gaseous radwaste systems holdup time (days)	11.3	45	45	45	45
System type		Charcoal delay	Compressed gas storage	Compressed gas storage	Charcoal delay
Solid radwaste solidification agent	11.4	Cement	Cement or vinyl ester styrene	Vermiculite-Portland cement	Cement

TABLE 1.3-3 (Sh. 1 of 4)

SIGNIFICANT DESIGN CHANGES

Item	Reference Section	Reason for Change
Technical Support Center (TSC)	Appendix 1B III.A.1.2	The location of TSC has been changed from a dedicated TSC for each unit to a common shared TSC in the access control bldg of unit 3 with a satellite TSC (STSC) in the computer room of unit 4.
Seismic Instrumentation	3.7.4.2	The mounting locations of peak accelerographs have been changed as part of the normal design development as follows: . PSAR a. on the accumulator tank in the containment building, b. on the safety injection piping in the containment building, and c. on the essential service return piping in the auxiliary building. . FSAR a. on the pressurizer inside the containment building, b. on the reactor coolant cold leg pipe inside the containment building, c. on the LPSI piping in the primary aux. building, and d. under the top of the dome near the polar crane.

TABLE 1.3-3 (Sh. 2 of 4)

Item	Reference Section	Reason for Change
Burnable Poison Rod	4.3	The core design has been changed from that stated in the PSAR by incorporating different enrichments and different number of burnable poison rods. There is no safety significance to this change.
Reactor Coolant Pressure Boundary Material	5.2	Inconel 600 material was originally specified for instrumentation and heater penetration nozzles installed on the pressurizer, RCS hot legs, and steam generator hot leg drains. This material has been changed to Inconel 690 to ensure the long term corrosion resistance in high temperature environments. There is no safety significance to this material change.
Safety Depressurization System	5.4.16	A safety grade system has been added to the design to depressurize the plant in the event of a total loss of feedwater and auxiliary feedwater. This event is considered to be beyond the current licensing design basis. Additional details on the safety depressurization system will be provided in an Amendment to this FSAR.
Containment Spray	6.1	The containment spray solution has been changed from sodium hydroxide (NaOH) to hydrazine to provide for improved iodine removal, better pH control, and elimination of alkali attack on containment equipment.

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TABLE 1.3-3 (Sh. 3 of 4)

Item	Reference Section	Reason for Change
Hydrogen Purge Filters	6.2.5	The hydrogen purge system is a backup non-safety-related system to the safety-related redundant hydrogen recombiners. The filters provided in this system are not required in order to meet the dose criteria of 10 CFR 100. For this reason, the hydrogen purge filters will not be tested in accordance with Regulatory Guide 1.52.
Auxiliary Feedwater System	6.3, 10.4.9	The maximum flow limitation of the auxiliary feedwater flow to the affected steam generators with or without a steam generator secondary pipe rupture has been increased from 2,000 gpm (7.57 m^3/min) and 1,320 gpm (5.00 m^3/min) to 2,300 gpm (8.71 m^3/min) and 1,550 gpm (5.87 m^3/min), respectively, in order to provide a more flexible operating range for selection of the restricting flow orifice provided in the AF supply header to each steam generator. There is no safety significance to this design change.
AAC System & Building	3.8, 8.3, 9.5.12	According to the requirements of 10 CFR 50.63 (July 1988) and Reg. Guide 1.155, one (1) AAC diesel generator and a dedicated seismic Category 1 AAC building have been added for both of the YGN 3&4 units to accommodate a postulated station blackout event. This change significantly improves the safety and reliability of the electrical distribution system in the plant.

TABLE 1.3-3 (Sh. 4 of 4)

Item	Reference Section	Reason for Change
Number of CCW Heat Exchangers	9.2.2.2	The number of CCW heat exchangers has been changed from one (1) per train to three (3) per train to enhance the heat exchanger maintenance flexibility as follows : When seawater temperature is below 31.7°C (89°F), only two (2) heat exchangers are required to operate, thus the third heat exchanger can be shutdown for maintenance without system operation interruption. 1
CVCS Reliability	9.3.4	Design improvement were made which improve charging reliability and preclude the possibility for gas binding.
AAC Diesel Generator Building HVAC	9.4.5.1	Due to addition of AAC DG building, HVAC system is added as a safety-related system.
Halon Suppression System	9.5.1.2.2	Deleted the halon suppression system (i.e. a part of the fire protection system) for the purpose of environmental protection. And the halon gas will be restricted by Montreal Declaration to prevent the environment pollution.
Provisions for Diverting Condenser Off-Gas to Containment	10.4.2	The provisions for diverting the radioactive condenser off-gas discharge from atmosphere to containment have been added to the condenser vacuum system to minimize release of contaminated condenser off-gas to atmosphere when high radiation level has been detected on the condenser off-gas exhaust line. The safety related containment isolation valves have been designed for containment isolation. There is no safety significance to this design.

1.4 IDENTIFICATION OF AGENTS AND CONTRACTORS

1.4.1 KEPCO's Qualifications and Experience

Korea Electric Power Corporation (KEPCO), the sole government organized power utility in Korea, has total responsibility for the construction and operation of YGN 3&4. KEPCO has a long history of building and operating conventional electric generating plants. The overall KEPCO organization is shown in Figure 13.1-1. The home office of KEPCO is in Seoul, the capital city of the Republic of Korea, about 300 miles (480 km) north-northeast of the site. KEPCO is the sole applicant for the construction permit and operating license for YGN 3&4.

KEPCO, as owner, is responsible for the design, construction, and operation of these units. KOPEC and Sargent & Lundy assist KEPCO in engineering, procurement, and construction management. KHIC, KAERI, ABB-Combustion Engineering, and General Electric Company supply the nuclear steam supply systems and the turbine generator. Korean Nuclear Fuel Company (KNFC), with the assistance of ABB-Combustion Engineering, will provide the nuclear fuel.

The Nuclear Power Construction Department (NPCD) of KEPCO is responsible for the construction of YGN 3&4; the Nuclear Safety Office (NSO) of KEPCO is responsible for the licensing affairs (C.P. and O.L.); and the Nuclear Power Generation Department (NPGD) of KEPCO is responsible for the safe operation and maintenance of the units in compliance with technical specifications and other applicable requirements. The organization of NPCD, NSO, 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 Architect-Engineers

Korea Power Engineering Company, Inc. (KOPEC), as a prime contractor, and Sargent & Lundy (S&L), as a subcontractor to KOPEC, have been retained by KEPCO to provide architect-engineering services and other related services for YGN 3&4 construction.

1.4.2.1 KOPEC's Qualifications and Experience

KOPEC was established in 1975 to meet the nation's increasing demands for architect-engineering capability, particularly in the field of electric power. Since its inception, KOPEC has played a leading role in consulting and engineering activities for all of the Korean nuclear projects and in providing various kinds of technical services for fossil, hydroelectric, and other energy-related facilities and structures.

KOPEC's nuclear experience dates back to 1976, when it undertook several design tasks on the Kori Nuclear Power Plant, Units 1 (587 MWe, PWR) and 2 (650 MWe, PWR). Subsequently, KOPEC has participated in the construction projects of all of Korea's nuclear power plants and has provided various engineering services while expanding its role.

Through Ulchin Nuclear Power Plant, Units 1 & 2, KOPEC has provided architect-engineering services for nuclear power plant construction for more than eight units with a total capacity of 6,900 MWe. In 1982, in recognition of the technical capability KOPEC had developed, the government and KEPCO designated KOPEC as the prime architect-engineering contractor for future nuclear power plants in Korea. KOPEC then started to execute the whole range of architect-engineering services for the design and construction of YGN 3&4. KOPEC has also started to participate in a new project for Wolsung Unit 2, as a subcontractor of AECL (Atomic Energy Canada Ltd.), and to execute the entire range of architect-engineering services for the design and construction of UCN

3&4, and also particpates in the Wolsung 3&4.

1.4.2.2 Sargent & Lundy's Qualifications and Experience

Sargent & Lundy (S&L) was established in Chicago, Illinois, U.S.A. in 1891. For over 100 years, S&L has specialized exclusively in the design of generation, transmission, distribution, and utilization of steam and electric power and related facilities. Throughout the years, S&L has designed generating capacity equivalent to approximately 20% of the U.S. privately owned investor central station generating capacity today, totaling nearly 90,000 MWe. S&L has designed more than 470 coal-fired units. Eighty-five coal-fired units of 200 MW or larger have been authorized for design, representing over 40,000 MWe of installed capacity. Sixty-seven of these units are in operation now.

S&L's contribution to nuclear power development includes 22 units totaling over 17,000 MWe. S&L has been responsible for the design of eight large PWR units. Table 1.4-1 lists the nuclear power plants designed by S&L.

1.4.3 NSSS Suppliers

YGN 3&4 is equipped with two-loop pressurized water reactor nuclear steam supply systems designed by the Joint System Design team of the Korea Atomic Energy Research Institute (KAERI) and ABB-Combustion Engineering, Inc. (ABB-CE), and manufactured by Korea Heavy Industries & Construction Co., Ltd. (KHIC) and ABB-CE.

The experience and qualifications of each entity are provided in the following subsections.

1.4.3.1 KHIC's Qualifications and Experience

KHIC was incorporated in 1962 under the name of Hyundai International Inc., with its main business lines as the manufacturer of various industrial machinery and equipment.

In the course of its development, the company changed management in November 1980 in compliance with the Korea Governmental Policy on Heavy Industry Distribution and was renamed "Korea Heavy Industries and Construction Co., Ltd." (KHIC).

This government action and subsequent transformation in the ownership of the company has strengthened KHIC's capabilities to assume sole responsibility for all nuclear power plants to be installed in Korea.

Under the above policy, KHIC has been designated by Korea Electric Power Corporation (KEPCO) as the prime contractor for the supply of equipment, materials, and related services and the installation of the nuclear steam supply system (NSSS) and turbine-generator (TG) for YGN 3&4.

In this project, KHIC's sole scope of supply includes component design and equipment manufacturing within NSSS and TG. In these fields, KHIC's capabilities are described in the following subsections.

1.4.3.1.1 Design and Engineering

KHIC's Engineering Division is capable of performing specific component design with the extensive support of KHIC's licensors, project collaborators, and engineering subcontractors. Depending upon specific requirements of the project, specialist consulting firms are utilized from time to time. At present, KHIC's qualified engineers are working on the engineering phase of the awarded project. KHIC's component design activities include material selection, thermo-fluid calculation, heat and mass balance, strength calculation, and shop detailing. External engineering assistance has been successfully utilized as demonstrated on successful past projects.

1.4.3.1.2 Manufacturing

The following partial list of items manufactured in KHIC's shop illustrates the firm's experience in supplying nuclear power plants.

a. Nuclear Steam Supply System

- reactor vessel
- steam generator
- pressurizer
- reactor coolant pump casing
- heat exchangers and tanks
- reactor coolant piping

b. Turbine and Accessories

- high-pressure turbine system
- low-pressure turbine system
- moisture separator-reheaters
- main steam valves
- main steam piping
- turning gear
- steam-sealing system
- lube oil system

c. Generator and Accessories

- generator
- excitation system
- hydrogen cooling and control system
- stator cooling water system

d. Balance of Plant

- condenser and accessories
- heat exchangers
- tanks and pressure vessels
- HVAC system
- intake facilities
- cranes
- evaporators
- pumps
- piping system (valves, hangers, etc.)

1.4.3.1.3 Major Facilities

The construction of the Changwon Plant started in late 1976 and was completed on June 29, 1982. This plant is the biggest manufacturing plant in Korea and one of the largest integrated plants in the world.

The plant includes machine, heavy machine, engine assembly and test shop, fabrication, heavy fabrication, steel foundry and forge shop, inspection and laboratory facilities, head office, and welfare facilities. It has the capability to manufacture the full range of machinery and equipment for nuclear power plants.

1.4.3.1.4 Quality Control

KHIC is dedicated to produce equipment in compliance with the applicable codes and standards and with clients' specifications.

KHIC has successfully performed work within the guidelines of numerous internationally recognized standards such as ASME, DIN, AWS, JIS, and TEMA.

KHIC's nondestructive examination (NDE) capability is among the best in Korea, and KHIC's NDE personnel are qualified in accordance with the requirements of SNT-TC-14.

1.4.3.1.5 Experience Records in Manufacturing

KHIC's experience in manufacturing is shown in Table 1.4-2.

1.4.3.2 KAERI's Qualifications and Experience

The Korea Atomic Energy Research Institute (KAERI) carries out broad programs concerned with developing indigenous nuclear power technology as well as performing fundamental research since 1959. KAERI is responsible for significant achievement in the academic development of nuclear area, and with a successful nuclear fuel design recently achieved localization of nuclear fuel. KAERI-made nuclear fuel has been loaded and successfully demonstrated in nuclear power plants. KAERI will secure the ability to be self-reliant in the nuclear fuel and NSSS system design field by developing reactor design technology.

1.4.3.2.1 R&D on Nuclear Fuel Cycle Technology

KAERI not only carries out domestically designed and manufactured nuclear fuel performance testing and evaluation, using the Hot Test Loop, Post-Irradiation

Examination Facilities and the planned MRR (Multipurpose Research Reactor) which will be constructed in the mid-1990s, but also conducts radioactive waste treatment and disposal projects. Through this process, KAERI is establishing the core technology of the nuclear fuel cycle.

The Radioactive Waste Treatment Facility handles various radioactive wastes produced in nuclear power plants and facilities. KAERI has completed the construction of a low- and intermediate-level radioactive waste treatment facility and the installation of research equipment. With this facility, KAERI will assume responsibility for the treatment of various radioactive wastes from Korean nuclear power plants and facilities.

1.4.3.2.2 R&D of Reactor Design

1.4.3.2.2.1 NSSS System Design Project

KAERI is an active participant in the nuclear power technology self-reliance program under the strong guidance of the government and mutual cooperation among power group industries: Korea Electric Power Corp. (KEPCO), Korea Heavy Industries & Construction Co. (KHIC), and Korea Power Engineering Co. Ltd. (KOPEC).

Standardization of nuclear power plants was conducted for better economy and technology transfer, and KAERI has been investigating and will be investing NSSS design improvements that may be applicable on future units.

1.4.3.2.2.2 Korea Multi-Purpose Research Reactor (KMRR)

The KMRR project has been and will be carried out with the utmost use of local technologies. Completion of the project will contribute to the establishment of a high-level nuclear industry. The KMRR conceptual design was completed at the end of 1985 and the detailed design at the end of 1991. Construction of the KMRR is planned to be completed by the end of 1993, with criticality to occur by the end of 1994.

The research reactor will be mainly used to test and develop locally fabricated materials and fuels for nuclear power reactors.

1.4.3.2.3 Nuclear Safety Research

To achieve the improvement of nuclear power plant safety and its availability in Korea, KAERI has placed emphasis upon the following safety-related research areas:

a. Operational safety of nuclear power plants
b. Severe accidents at nuclear power plants
c. Environmental safety of nuclear facilities
d. Health effects of radiation

Nondestructive testing and pressure vessel surveillance testing have been introduced and localized to promote the operational safety of nuclear power plants.

1.4.3.2.4 Research Reactor

The first research reactor in Korea is the TRIGA Mark-II, which has been operating since 1962. The reactor was designed to operate at the thermal power of 100 kW, which was locally upgraded to 250 kW in 1968. The second research reactor, TRIGA Mark-III, was completed in 1972 and has been operated since then. The research reactors produce a large number of radioisotopes for industrial, agricultural, and medical applications and supply about 14 kinds of labeled compounds.

1.4.3.2.5 Development of Radiation Application Technology

In the broad radiation application spectrum, KAERI has emphasized the production of radioisotopes/labeled compounds, radiation processing, food preservation, and radiation mutation studies.

More than 10 radionuclides and 14 labeled compounds, including the extraction-type 99mTc generator and radioimmunoassay kits, are now in routine production. In accordance with the integrated long-term program, KAERI is now scaling up radioisotope production using both a medical cyclotron and the forthcoming multipurpose, high neutron flux research reactor.

1.4.3.2.6 Nuclear Training Center

KAERI founded the Nuclear Training Center in 1968 in order to carry out in-house training for KAERI employees as well as open-door training for applicants from nuclear-related organizations. Since 1983, the Nuclear Training Center has been empowered by the government authorities to examine and license management associated with reactor operators, special nuclear materials handlers, radiosotope handlers, and other licensees. Typical training courses include the following:

a. Nuclear Power Generation Technology and Nuclear Materials Technology

- Plant Systems and Major Components and their Functions
- Safety Analysis
- Quality Control, Quality Assurance, and Inspection Technology
- Law, Decrees, Regulations, and Codes and Standards
- Nuclear Fuel Technology and Fuel Cycle
- Nuclear Emergency Preparedness

b. The Application of Radioisotopes and Radiation

c. Nondestructive Testing and Welding Technology

d. Retraining of Various Licensees in the Nuclear Sector

e. Reactor Experiments for College Students

1.4.3.3 ABB-Combustion Engineering's Qualifications and Experience

ABB-Combustion Engineering Nuclear Power, Inc., hereafter referred to as ABB-CE or Combustion Engineering, deals in nuclear power activities of three general types; design, development, construction and operation of reactor and auxiliary systems; design and fabrication of nuclear components; and support of design, development, and analytical projects.

A summary of the company's efforts, accomplishments, and operating experience in the light water reactor field is provided below.

1.4.3.3.1 Precommercial Reactor Programs

1.4.3.3.1.1 Naval Propulsion Program

During the period 1955 through 1960, ABB-CE was a major contributor to the U.S. Naval Reactors program. The company designed and built at its Windsor, Connecticut site, the prototype of a small attack submarine power plant. This prototype (S1C) went into operation in 1959 and currently operates as a naval training facility. A second plant of this type was also designed and built by ABB-CE for installation in the USS Tullibee (SSN-597), which operated as a part of the United States nuclear submarine fleet.

In the design, development, construction, and operation of the prototype system and the submarine power plant, ABB-CE's responsibilities included all safety aspects of the reactor systems.

1.4.3.3.1.2 Boiling Nuclear Superheat Plant

ABB-CE was responsible for the nuclear design and for the direction of startup and initial operation of the boiling nuclear superheat (BONUS) plant in Puerto Rico.

The design of this reactor system presented a number of unique problems, e.g., control and safety analysis of a two-region core, design of a superheater fuel element, design of a steam control system to ensure adequate cooling of superheat fuel under all credible conditions, and design of a containment building of the "total containment" type to house the entire power generating installation.

The BONUS plant achieved full power operation in September 1965 and was the first nuclear power plant under U.S.AEC control operating with an integral superheating core.

1.4.3.3.2 Development and Design of Commercial PWR Systems

The development and design by ABB-CE of a pressurized water reactor for utility service dates back to 1958. At that time, the company was selected by the AEC to undertake the design, analysis, and economic evaluation of a 250-MWe PWR plant, in conjunction with an architect-engineer. This effort provided initial technical and economic guidelines for ABB-CE's commercial development of the PWR.

With a subsequent decision by ABB-CE to concentrate on the development of the PWR for large nuclear power stations, a program was initiated to guide required design and development work along appropriate lines. The following is representative of the types of PWR-oriented work that have been performed:

a. Evaluation of overall plant and systems to establish optimum physical arrangement and design criteria from the standpoint of economics and safety, much of this in conjunction with qualified architect-engineering organizations

b. Design and development of nuclear components such as control assemblies, control assembly drive mechanisms, and auxiliary systems equipment

c. Extensive testing of PWR nuclear components, such as fuel assemblies and reactor control components, under actual service pressure, temperature, and flow conditions

For many years, ABB-CE's Nuclear Laboratories have been engaged in the development and testing of fuels, fuel elements, control assemblies, reactor components, and materials for reactor application. Particular emphasis has been given to UO_2 and zircaloy cladding technology, involving both in-pile and out-of-pile investigations. The initial efforts in the laboratories were associated with submarine reactor programs. Since 1960, the personnel of the Nuclear Laboratories have actively participated in the joint U.S. AEC-Euratom research and development program for fuels development. In addition to these programs, personnel in the Nuclear Laboratories have been responsible for materials design activities for the heavy water organically-cooled reactor study and for pressurized water, boiling water, nuclear superheat, and fast breeder reactor systems.

1.4.3.3.3 Major Component Design and Fabrication

During the period of 1955-1961, ABB-CE was a major supplier of nuclear cores for naval propulsion service. The company fabricated the boiling and the superheating fuel for the BONUS reactor. The boiling section of the BONUS core was made up of zircaloy clad, rod type, UO_2 fuel elements fundamentally similar to those being utilized in the ABB-CE standard fuel design. The BONUS superheater fuel utilized Inconel-clad rod type UO_2 fuel elements. The superheater cladding was designed for an operating temperature of 1250°F (676.7°C).

ABB-CE has performed the design engineering and fabrication of control rod drive mechanisms and fuel rods for all of the commercial power reactors listed in Table 1.4-3 at its facilities in Windsor and East Windsor, Connecticut.

ABB-CE has fabricated and shipped many reactor vessels for utility plant service and for naval service.

The company supplied nuclear steam generators for naval service and for all PWR plants listed in Table 1.4-3. In addition, ABB-CE company designed and fabricated ten steam generators installed in the Hanford New Production Reactor facility.

Processing of low enrichment uranium dioxide and manufacture into high-quality nuclear fuel bundles occurs at the ABB-CE Windsor and Hematite nuclear fuel fabrication facilities. This fuel has been supplied as initial and reload cores to many U.S.A. PWRs.

ABB-CE Newington, a subsidiary of ABB-CE, is a highly experienced organization with facilities for manufacturing all reactor vessel internal structures.

ABB-CE extended its manufacturing capability to include fabrication of reactor coolant pumps by its entry in 1974 into joint ownership, as the majority owner, of the CE/KSB Pump Company.

1.4.3.3.4 Facilities

The CE laboratories at Windsor, Connecticut, and Chattanooga, Tennessee, provide complete facilities for the development, design, analysis and testing of PWR components and systems. These laboratories include equipment for the following

a. Mechanical testing

b. X-ray and radiography

c. Metallography

d. Ceramics development

e. Analytical and radio-chemistry

f. Fuel fabrication development

g. Corrosion testing

h. 2500 psi (175.75 kg/cm^2) component performance testing

i. 2500 psi (175.75 kg/cm^2) and 5000 psi (351.5 kg/cm^2) steam generation

j. Welding development

The Windsor facilities of ABB-CE are equipped to fabricate and provide the necessary quality control for fuel assemblies, control assemblies, control assembly drive mechanisms, and other specialized nuclear components.

ABB-CE's Chattanooga Plant is equipped to design, fabricate, and provide quality control for large reactor pressure components. The facility has special equipment such as heavy-duty cranes and large-capacity machine tools capable of performing work on large, heavy parts to close tolerances and fine surface finishes. It is also equipped with the latest testing and quality control equipment, including a linear accelerator for weld examination.

1.4.3.3.5 Commercial Reactor Operation

Table 1.4-3 lists plants with pressurized water reactor nuclear steam supply systems designed and manufactured by ABB-CE.

1.4.3.3.5.1 Palisades

Palisades is the oldest operating commerical electric generating facility having a ABB-CE Nuclear Steam Supply System. It achieved commercial operation in December 1971. Operated by Consumers Power Company, this plant has a design net electrical rating of 805 MWe.

1.4.3.3.5.2 Maine Yankee

Maine Yankee Nuclear Power Plant is the second-oldest plant supplied with a ABB-CE Nuclear Steam Supply System; it entered commercial operation in December 1972. This three-loop 830-MWe plant is owned and operated by the Maine Yankee Atomic Power Company.

1.4.3.3.5.3 Fort Calhoun

The Fort Calhoun Nuclear Plant entered commercial operation in June 1974. This 478-MWe plant is operated by the Omaha Public Power District.

1.4.3.3.5.4 Millstone Point Unit 2

The 870-MWe Millstone Point Unit 2 plant entered commercial operation in December 1975. The plant is operated by the Northeast Nuclear Energy Company.

1.4.3.3.5.5 Calvert Cliffs Units 1 and 2

Calvert Cliffs Units 1 and 2 achieved commercial operation in May 1975 and April 1977, respectively. These 845-MWe units are operated by Baltimore Gas and Electric.

1.4.3.3.5.6 St. Lucie Units 1 and 2

The 830-MWe St. Lucie nuclear plants commenced commercial operation in December 1976 and August 1983, respectively, for Units 1 and 2. St. Lucie Unit 2 required only 6 years for construction and licensing. These plants are operated by the Florida Power & Light Company.

1.4.3.3.5.7 Arkansas Nuclear One Unit 2

The Arkansas Nuclear One Unit 2 912-MWe plant entered commercial operation in March 1980. It is operated by Arkansas Power & Light Co.

1.4.3.3.5.8 San Onofre Units 2 and 3

San Onofre Units 2 and 3 began commercial operation in August 1983 and April 1984, respectively. These units have a net design rating of 1070 and 1080 MWe, respectively, and are operated by Southern California Edison Company.

1.4.3.3.5.9 Waterford Unit 3

The 1104-MWe Waterford Unit 3 is operated by Louisiana Power & Light Company. Commercial operation was achieved in September 1985.

1.4.3.3.5.10 Palo Verde Units 1, 2, and 3

The three 1270-MWe Palo Verde units represent the first nuclear plants licensed in the United States using a standardized reference design (System 80™). Units 1 and 2 achieved commercial operation in January and September 1986, respectively; Unit 3 began commercial operation in January 1988. The plants are operated by Arizona Public Service Company.

1.4.3.3.5.11 Washington Nuclear Project Unit 3

Construction of Washington Nuclear Project Unit 3 is nearly 80% complete. The constructing utility, Washington Public Power Supply System, has delayed completion of this plant. The NSSS is a 1240-MWe System 80™ design.

1.4.4 Turbine-Generator Supplier

The turbine-generators for YGN 3&4 are designed by General Electric (GE) and jointly manufactured by GE and KHIC. KHIC will manufacture its scope of supply to manufacturing information supplied by GE, with the assistance of a resident GE manufacturing liaison engineer.

The qualifications and experience of KHIC and GE are given in the following.

1.4.4.1 KHIC's Qualifications and Experience

Refer to Subsection 1.4.3.1.

1.4.4.2 GE's Qualifications and Experience

GE has supplied more than 80 turbine-generators to nuclear power plants.

GE's experience in supplying turbine-generators to nuclear power plants is listed in Table 1.4-4.

TABLE 1.4-1 (Sh. 1 of 2)

NUCLEAR UNITS AUTHORIZED FOR

DESIGN BY SARGENT & LUNDY

CLIENT	STATION-UNIT	TYPE OF REACTOR*	MW	YEAR OF COMMERCIAL OPERATION
Commonwealth Edison Company	Dresden 2	BWR	850	1970
	Dresden 3	BWR	850	1971
	Quad-Cities 1	BWR	850	1972
	Quad-Cities 2	BWR	850	1972
	Zion 1	PWR	1130	1973
	Zion 2	PWR	1130	1974
	La Salle 1	BWR	1170	1982
	La Salle 2	BWR	1170	1984
	Byron 1	PWR	1175	1985
	Byron 2	PWR	1175	1987
	Braidwood 1	PWR	1175	1987
	Braidwood 2	PWR	1175	1988
The Cincinnati Gas & Electric Company	Zimmer	BWR	870	Canceled**
Dairyland Power Cooperative	La Crosse	BWR	50	1969
Illinois Power Company	Clinton 1	BWR	985	1986
Northern Indiana Public Service Company	Bailly N1	BWR	697	Canceled**
Public Service Company of Colorado	Fort St. Vrain 1	HTGR	341	1979
Public Service Indiana	Marble Hill 1	PWR	1175	Canceled**
	Marble Hill 2	PWR	1175	Canceled**

* BWR - boiling water reactor
HTGR - high temperature gas reactor
LMFBR - liquid metal fast breeder reactor
PWR - pressurized water reactor

** Substantial design work had been completed before the project was canceled.

TABLE 1.4-1 (Sh. 2 of 2)

CLIENT	STATION-UNIT	TYPE OF REACTOR*	MW	YEAR OF COMMERCIAL OPERATION
Southwest Atomic Energy Associates	SEFOR	LMFBR	No turbine	1967
United Power Association	Elk River	BWR	No turbine	1961
U.S. Atomic Energy Commission	Borax 14	BWR	3	1955
	EBWR	BWR	5	1956

TABLE 1.4-2

KHIC NUCLEAR UNIT MANUFACTURING EXPERIENCE

CLIENT	PROJECT NAME	LOCATION	DESCRIPTION	PERIOD
KEPCO	Kori 3&4	Kori, Korea	▪ Auxiliary boiler ▪ Water intake facility	08/81-11/83
W. J. Wooley			▪ Missile-resistant door	09/82-08/83
VSL Corp.			▪ Post-tensioning system	12/79-01/84
W/H	YGN 1&2	Yonggwang, Korea	▪ NSSS & TG	06/81-03/85
KEPCO			▪ Auxiliary boiler ▪ Turbine crane ▪ Water screen water Chiller ▪ Structural steel	06/81-10/84
Southwest Eng.			▪ FW heater ▪ Condenser, Hx.	05-81-09/84
Ederer Inc.			▪ Polar crane	08/81-06/83
AAF Co.			▪ Air handling units	12/81-07/84
W. J. Wooley			▪ Airlock and hatch	02/82-04/83
VSL Corp.			▪ Post-tensioning system	02/81-01/84
KEPCO	Ulchin 1&2	Ulchin, Korea	▪ CLP and SLP piping ▪ Gantry crane ▪ MWTS, SWS, SS	12/82-04-86
Framatome			▪ Nuclear island	12/81-04/86
Alsthom			▪ Conventional island	04/82-09/86
Neyrpic SA			▪ Personnel airlocks	04/85-09/86

TABLE 1.4-3

ABB-CE PRESSURIZED WATER REACTOR PLANTS

PLANT	OPERATOR UTILITY	PLANT LOCATION	COMMERCIAL OPERATION	DESIGN MWe NET
Non-System 80 Plants				
Palisades	Consumers Power Co.	Michigan	1971	805
Fort Calhoun	Omaha Public Power District	Nebraska	1974	478
Maine Yankee	Yankee Atomic Power Co.	Maine	1972	840
Calvert Cliffs 1	Baltimore Gas & Electric Co.	Maryland	1975845	
Calvert Cliffs 2	Baltimore Gas & Electric Co.	Maryland	1977845	
St. Lucie 1	Florida Power & Light Co.	Florida	1976	830
St. Lucie 2	Florida Power & Light Co.	Florida	1983	870
Millstone Point 2	Northeast Utilities	Connecticut	1975	1070
San Onofre 2	Southern California Edison Co.	California	1983	1080
San Onofre 3	Southern California Edison Co.	California	1984	912
Arkansas Nuclear One 2	Arkansas Power & Light Co.	Arkansas	1980	1075
Waterford 3	Louisiana Power & Light Co.	Louisiana	1985	830
System 80 Plants				
Palo Verde 1	Arizona Public Service Company	Arizona	1986	1270
Palo Verde 2	Arizona Public Service Company	Arizona	1986	1270
Palo Verde 3	Arizona Public Service Company	Arizona	1988	1270
WNP-3	Washington Public Power Supply System	Washington	N/A	1240

TABLE 1.4-4 (Sh. 1 of 4)

NUCLEAR POWER PLANTS WITH GENERAL ELECTRIC TURBINE-GENERATOR UNITS

UTILITY	STATION	RATING (MWe)	IN SERVICE (YEAR)
Commonwealth Edison	Dresden 1	192	1960
	Dresden 2	810	1970
	Dresden 3	810	1971
	Quad-Cities 1	810	1972
	Quad-Cities 2	810	1972
	La Salle 1	1147	1982
	La Salle 2	1147	1984
Washington Public Power System	Hanford Sta. 1-1	422	1966
	Hanford Sta. 1-2	422	1966
GPU Nuclear Corp.	Oyster Creek 1	640	1969
	TMI-1	837	1974
India Atomic Power	Tarapur-1	215	1969
	Tarapur-2	215	1969
Niagara Mohawk	Nine Mile Point 1	620	1969
	Nine Mile Point 2	1166	1986
Atomic Power Corp.	Tsuruga 1	357	1969
	Tokai 2	1110	1978
Northeast Utility	Millstone Point 1	650	1970
	Millstone Point 2	880	1975
	Millstone Point 3	1208	1986
Tokyo Electric Power Co.	Fukushima 1	461	1970
	Fukushima 2	783	1973
	Fukushima 6	1110	1979
Spain Nuclenor	Garona 1	460	1971
Northern States Power	Monticello 1	542	1971
Boston Edison	Pilgrim 1	655	1972
Vermont Yankee Nuclear Power	Vermont Yankee	537	1972

TABLE 1.4-4 (Sh. 2 of 4)

UTILITY	STATION	RATING (MWe)	IN SERVICE (YEAR)
Duke Power	Oconee 1	886	1973
	Oconee 2	886	1973
	Oconee 3	893	1974
	Catawba 1	1205	1985
	Catawba 2	1205	1986
Omaha Public Power District	Fort Calhoun	481	1973
Tennessee Valley Authority	Browns Ferry 1	1098	1973
	Browns Ferry 2	1098	1974
	Browns Ferry 3	1091	1976
Philadelphia Electric	Peach Bottom 2	1098	1974
	Peach Bottom 3	1098	1974
	Limerick 1	1092	1985
	Limerick 2	1092	1990
Baltimore Gas & Electric	Calvert Cliffs 1	890	1974
Iowa Electric Lighting & Power	Arnold 1	565	1974
Georgia Power	Hatch 1	809	1974
	Hatch 2	820	1978
	Vogtle 1	1160	1986
	Vogtle 2	1160	1988
Indiana & Michigan Electric Co.	Cook 1	1089	1975
New York Power Authority	Fitzpatrick 1	849	1975
Carolina Power & Light	Brunswick 2	849	1975
	Brunswick 1	849	1976
Portland General Electric Company	Trojan 1	1177	1975
Public Service of Colorado	Fort St. Vrain 1	336	1976
Toledo Edison	Davis Besse 1	925	1977
Arkansas Power & Light	Arkansas Nuc. 2	942	1978

TABLE 1.4-4 (Sh. 3 of 4)

UTILITY	STATION	RATING (MWe)	IN SERVICE (YEAR)
Pennsylvania Power & Light	Susquehanna 1	1084	1982
	Susquehanna 2	1084	1984
South Carolina Electric & Gas Co.	Summer 1	953	1982
Hydro Quebec	Gentilly 2	685	1982
Taipower	Maanshan 1	951	1984
	Maanshan 2	951	1985
Ontario Hydro	Bruce B (660)	807	1984
	Bruce B (560)	807	1984
	Bruce B (760)	807	1986
	Bruce B (860)	807	1986
Hidroelectrica Espanola SA (Spain)	Cofrentes 1	974	1984
Union Electric Co.	Callaway 1	1192	1984
Ministry of Electric Energy (Romania)	Cernavoda 1	706	1985
	Cernavoda 2	706	1987
Kansas Gas & Electric	Wolf Creek 1	1192	1985
Arizona Public Service	Palo Verde 1	1359	1985
	Palo Verde 2	1359	1986
	Palo Verde 3	1359	1986
Long Island Lighting Co.	Shoreham 1	846	1985
Gulf States Utility	River Bend 1	997	1985
Public Service of New Hampshire	Seabrook 1	1197	1986
Illinois Power Company	Clinton Power 1	984	1986
Cleveland Electric Illuminating	Perry 1	1252	1986
	Perry 2	1252	1988

TABLE 1.4-4 (Sh. 4 of 4)

UTILITY	STATION	RATING (MWe)	IN SERVICE (YEAR)
Cent Nuclear (Spain)	Valdecaballeros 1	974	1986
	Valdecaballeros 2	974	1999
Public Service Electric & Gas	Hope Creek 1	1117	1986

1.5 REQUIREMENTS FOR FURTHER TECHNICAL INFORMATION

The YGN 3&4 NSSS design represents a combination of previously approved design methods. As such, there is no development program currently underway that is necessary to support construction or operation.

1.5.1 Reactor Flow Model Testing

A scale flow model of the YGN 3&4 reactor vessel and internals was tested. A detailed discussion of the flow model and test facility appears in Section 4.4. The purpose of this test was to confirm the reactor hydraulic characteristics, thus verifying the design methods employed in downsizing from large to small System 80. In particular, core inlet flow distributions and pressure losses along the flow path segments within the reactor vessel were determined based upon the test results. The YGN 3&4 flow model tests was performed in 1989. Results from this test program were incoporated into the thermal-hydraulic design analysis.

1.6 MATERIAL INCORPORATED BY REFERENCE

The following list is a tabulation of all material incorporated by reference as part of this application. Other material for information purposes, not incorporated by reference, is listed in the individual chapter and section references.

Report No.	Title	Date Issued	FSAR Chapter
CENPD-67	Combustion Engineering, Inc. "Iodine Decontamination Factors During PWR Steam Generation and Steam Venting"	September 1973	10
Suppl. #1		May 1974	11
Suppl. #2		June 1974	
Addendum 1		November 1974	
Addendum 2		August 1975	
CENPD-80	Moisture Carryover During an NSSS Steamline Break Accident	January 1973	6
CENPD-98-A	COAST Code Description	April 1975	4 5 15
CENPD-118	Combustion Engineering, Inc. "Densification of Combustion Engineering Fuel"	June 1974	4
CENPD-133	Combustion Engineering, Inc. "CEFLASH-4A FORTRAN IV Digital Computer Program for Reactor Blowdown Analysis	August 1974	6
Suppl. #1	CEFLASH-4AS, A Computer Program for Reactor Blowdown Analysis of the Small Break Loss-of-Coolant Accident	September 1974	
Suppl. #2	CEFLASH-4A, A FORTRAN IV Digital Computer Program for Reactor Blowdown Analysis (Modifications)	March 1975	
Suppl. #3		February 1977	
Suppl. #4-P	CEFLASH-4A, A FORTRAN IV Digital Computer Program for Reactor Blowdown Analysis	April 1977	
Suppl. #5-P	CEFLASH-4A, A FORTRAN 77 Digital Computer Program for Reactor Blowdown Analysis	June 1985	

Report No.	Title	Date Issued	FSAR Chapter
CENPD-134	Combustion Engineering, Inc. "COMPERC-II A Program for	August 1974	6
Suppl. #1	Emergency Refill - Reflood of	February 1975	
Suppl. #2	the Core"	June 1985	
CENPD-135	Combustion Engineering, Inc.	August 1974	6
Suppl. #2	"STRIKIN-II A Cylindrical	February 1975	
Suppl. #4	Geometry Fuel Rod Heat	August 1976	
Suppl. #5	Transfer Program"	April 1977	
CENPD-137	Combustion Engineering, Inc. "Calculative Methods for the	August 1974	6
Suppl. #1	C-E Small Break LOCA Evaluation Model"	January 1977	
CENPD-138	PARCH - A FORTRAN IV	August 1974	6
Suppl. #1	Digital Computer Program to	February 1975	
Suppl. #2	Evaluate Pool - Boiling Axial Rod, and Coolant Heatup	January 1977	
CENPD-139-P-A	C-E Fuel Evaluation Model	July 1974	6
CENPD-162-A	Combustion Engineering, Inc. "CHF Correlation for C-E Fuel	September 1976	4
Suppl. #1-A	Assemblies with Standard Spacer Grids - Part 1; Uniform Axial Power Distribution"	February 1977	
CENPD-170	Combustion Engineering, Inc.	August 1975	4
Suppl. #1"	Assessment of the Accuracy of the PWR Safety System Actuation as Performed by the Core Protection Calculators"	November 1975	
CENPD-179	Combustion Engineering, Inc. "C-E Thermo-Structural Fuel Evaluation Method"	April 1976	4
CENPD-180	Radioiodine Behavior in Reactor	March 1976	15
Suppl. #1	Coolant During Transient Operation	March 1977	
CENPD-182	Combustion Engineering, Inc.	November 1975	3
Rev. 1	"Seismic Qualification of C-E Instrumentation and Electrical Equipment"	June 1977	

Report No.	Title	Date Issued	FSAR Chapter
CENPD-183-A	Combustion Engineering, Inc. "C-E Methods for Loss of Flow Analysis"	August 1975	15
CENPD-187-A	Combustion Engineering, Inc. "Method of Analyzing Creep	March 1976	4
Suppl. #1-A	Collapse of Oval Cladding"	June 1977	
CENPD-188-A	HERMITE, A Multi-Dimensional Space-Time Kinetics Code for PWR Transients	March 1976	4
CENPD-190	Combustion Engineering, Inc. "C-E Method for Control Element Assembly Ejection Analysis"	January 1976	15
CENPD-198	Combustion Engineering, Inc. "Zircaloy Growth-In-Reactor Dimensional Changes in Zircaloy-4 Fuel Assemblies"	December 1975	4
CENPD-201-A Suppl. #1	Reactor Coolant Pump Performance	April 1976 January 1981	5
CENPD-206-A	Combustion Engineering, Inc. "TORC Code Verification and Simplified Modeling Method"	June 1981	4
CENPD-207-A	Combustion Engineering, Inc. "Critical Heat Flux Correlation for C-E Fuel Assemblies with Standard Spacer Grids, Part 2, Non-Uniform Axial Power Distributions"	December 1984	4
CENPD-210-A Rev. 7	Quality Assurance Program A Description of the C-E Nuclear Power Businesses Quality Assurance Program	April 1992	17
CENPD-213 Suppl. #1	Combustion Engineering, Inc. "Application of FLECHT Reflood Heat Transfer Coefficients to Combustion Engineering 16 x 16 Fuel Bundles"	January 1976 March 1976	6

Report No.	Title	Date Issued	FSAR Chapter
CENPD-225-A	Combustion Engineering, Inc. "Fuel and Poison Rod Bowing"	June 1983	4
CENPD-254-A	"Post-LOCA Long-Term Cooling Evaluation Model"	June 1977	6
CENPD-255-A	"Qualification of Class 1E Instrumentation"	October 1981	7
CESSAR PSAR	Appendix 6B.		
Enclosure 1-P to LD-81-095	"C-E ECCS Evaluation Model Flow Blockage Analysis"	December 1981	6
Enclosure 1-P to LD-82-001	"CESEC: Digital Simulation of a Combustion Engineering Nuclear Steam Supply System"	January 1982	15
CENPD-132-P Suppl. #1 Suppl. #2 Suppl. #3-P	"Calculative Methods for the C-E Large Break LOCA Evaluation Model"	August 1974 February 1975 July 1975 June 1985	6
CENPD-161	"TORC Code: A Computer Code for Determining the Thermal Margin of the Reactor Core"	July 1975	4
CENPD-266-A	"The ROCS and DIT Computer Codes for Nuclear Design"	April 1983	4
CENPD-269 Rev. 1	"Extended Burnup Operation of Combustion Engineering PWR Fuel"	July 1984	4
CENPD-275	"C-E Methodology for Core Designs Containing Gadolinia Urania Burnable Absorbers"	March 1987	4

1.7 DRAWINGS AND OTHER DETAILED INFORMATION

Figures listed in Table 1.7-1, 1.7-2 and 1.7-3 are typical and the only controlled drawings are actually used to operate the plants. The others included in FSAR are controlled whenever they are revised. The list includes FSAR Figure & section number and drawing type, number, title, revision number, revision date of controlled drawing. 499

1.7.1 Electrical and Instrumentation and Control Drawings

The following categories of safety-related electrical drawings are listed in Table 1.7-1, in accordance with USNRC Regulatory Guide 1.70, Revision 3.

a. One-line diagrams
b. Key diagrams
c. Relay and metering drawings
d. Schematic drawings
e. Cable tray layout drawings
f. Control and instrumentation diagrams (C&ID)
G. Control logic diagrams (CLD)

1.7.2 Piping and Instrumentation Diagrams

The diagrams are listed in Table 1.7-2.

1.7.3 Other Diagrams

The diagrams are listed in Table 1.7-3.
a. General Arrangement Diagrams(GA)
b. Room Numbering Design Base(RN)
c. 345kV Switchyard Equipment Plan(E&ID) : Non Safety-related 499
d. Diagrams related to Fire Barrier & Protection Design Base(FB & FP)
e. Radiation Zones(RZ)
f. Main Control Board Elevation Diagrams(MCBE)
g. others(etc.)

1.7.4 Other Detailed Information

Additional information incorporated in this FSAR, if requested by the NSSC, will be described in this section. 582

TABLE 1.7-1(Sh. 1 of 54)
Electrical and Instrumentation and Control Drawings

FSAR Figure No. [(a)]	FSAR Section No.	Drawing Type	Controlled Drawing No.	Controlled Drawing Title	Revision No.	Revision

TABLE 1.7-1(Sh. 2 of 54)
Electrical and Instrumentation and Control Drawings

FSAR Figure No. (a)	FSAR Section No.	Drawing Type	Controlled Drawing No.	Controlled Drawing Title	Revision No.	Revision

TABLE 1.7-1(Sh. 3 of 54)
Electrical and Instrumentation and Control Drawings

FSAR Figure No. (a)	FSAR Section No.	Drawing Type	Controlled Drawing No.	Controlled Drawing Title	Revision No.	Revision

TABLE 1.7-1(Sh. 4 of 54)
Electrical and Instrumentation and Control Drawings

FSAR Figure No. [a]	FSAR Section No.	Drawing Type	Controlled Drawing No.	Controlled Drawing Title	Revision No.	Revision

TABLE 1.7-1(Sh. 5 of 54)
Electrical and Instrumentation and Control Drawings

FSAR Figure No. [(a)]	FSAR Section No.	Drawing Type	Controlled Drawing No.	Controlled Drawing Title	Revision No.	Revision

TABLE 1.7-1(Sh. 6 of 54)
Electrical and Instrumentation and Control Drawings

FSAR Figure No. [a]	FSAR Section No.	Drawing Type	Controlled Drawing No.	Controlled Drawing Title	Revision No.	Revision

TABLE 1.7-1(Sh. 7 of 54)
Electrical and Instrumentation and Control Drawings

FSAR Figure No. [(a)]	FSAR Section No.	Drawing Type	Controlled Drawing No.	Controlled Drawing Title	Revision No.	Revision

TABLE 1.7-1(Sh. 8 of 54)
Electrical and Instrumentation and Control Drawings

FSAR Figure No. [a]	FSAR Section No.	Drawing Type	Controlled Drawing No.	Controlled Drawing Title	Revision No.	Revision

TABLE 1.7-1(Sh. 9 of 54)
Electrical and Instrumentation and Control Drawings

FSAR Figure No.[(a)]	FSAR Section No.	Drawing Type	Controlled Drawing No.	Controlled Drawing Title	Revision No.	Revision

TABLE 1.7-1(Sh. 10 of 54)
Electrical and Instrumentation and Control Drawings

FSAR Figure No. [(a)]	FSAR Section No.	Drawing Type	Controlled Drawing No.	Controlled Drawing Title	Revision No.	Revision

TABLE 1.7-1(Sh. 11 of 54)
Electrical and Instrumentation and Control Drawings

FSAR Figure No. [(a)]	FSAR Section No.	Drawing Type	Controlled Drawing No.	Controlled Drawing Title	Revision No.	Revision

TABLE 1.7-1(Sh. 12 of 54)
Electrical and Instrumentation and Control Drawings

FSAR Figure No. (a)	FSAR Section No.	Drawing Type	Controlled Drawing No.	Controlled Drawing Title	Revision No.	Revision

TABLE 1.7-1(Sh. 13 of 54)
Electrical and Instrumentation and Control Drawings

FSAR Figure No. (a)	FSAR Section No.	Drawing Type	Controlled Drawing No.	Controlled Drawing Title	Revision No.	Revision

TABLE 1.7-1(Sh. 14 of 54)
Electrical and Instrumentation and Control Drawings

FSAR Figure No. (a)	FSAR Section No.	Drawing Type	Controlled Drawing No.	Controlled Drawing Title	Revision No.	Revision

TABLE 1.7-1(Sh. 15 of 54)
Electrical and Instrumentation and Control Drawings

FSAR Figure No. (a)	FSAR Section No.	Drawing Type	Controlled Drawing No.	Controlled Drawing Title	Revision No.	Revision

TABLE 1.7-1(Sh. 16 of 54)
Electrical and Instrumentation and Control Drawings

FSAR Figure No. [(a)]	FSAR Section No.	Drawing Type	Controlled Drawing No.	Controlled Drawing Title	Revision No.	Revision

TABLE 1.7-1(Sh. 17 of 54)
Electrical and Instrumentation and Control Drawings

FSAR Figure No. [a]	FSAR Section No.	Drawing Type	Controlled Drawing No.	Controlled Drawing Title	Revision No.	Revision

TABLE 1.7-1(Sh. 18 of 54)
Electrical and Instrumentation and Control Drawings

FSAR Figure No. [(a)]	FSAR Section No.	Drawing Type	Controlled Drawing No.	Controlled Drawing Title	Revision No.	Revision
[redacted]						

TABLE 1.7-1(Sh. 19 of 54)
Electrical and Instrumentation and Control Drawings

FSAR Figure No.[(a)]	FSAR Section No.	Drawing Type	Controlled Drawing No.	Controlled Drawing Title	Revision No.	Revision

TABLE 1.7-1(Sh. 20 of 54)
Electrical and Instrumentation and Control Drawings

FSAR Figure No. [(a)]	FSAR Section No.	Drawing Type	Controlled Drawing No.	Controlled Drawing Title	Revision No.	Revision

TABLE 1.7-1(Sh. 21 of 54)
Electrical and Instrumentation and Control Drawings

FSAR Figure No. [(a)]	FSAR Section No.	Drawing Type	Controlled Drawing No.	Controlled Drawing Title	Revision No.	Revision

TABLE 1.7-1(Sh. 22 of 54)
Electrical and Instrumentation and Control Drawings

FSAR Figure No. [(a)]	FSAR Section No.	Drawing Type	Controlled Drawing No.	Controlled Drawing Title	Revision No.	Revision

TABLE 1.7-1(Sh. 23 of 54)
Electrical and Instrumentation and Control Drawings

FSAR Figure No.[(a)]	FSAR Section No.	Drawing Type	Controlled Drawing No.	Controlled Drawing Title	Revision No.	Revision

TABLE 1.7-1(Sh. 24 of 54)
Electrical and Instrumentation and Control Drawings

FSAR Figure No. [(a)]	FSAR Section No.	Drawing Type	Controlled Drawing No.	Controlled Drawing Title	Revision No.	Revision Date

TABLE 1.7-1(Sh. 25 of 54)
Electrical and Instrumentation and Control Drawings

FSAR Figure No. [(a)]	FSAR Section No.	Drawing Type	Controlled Drawing No.	Controlled Drawing Title	Revision No.	Revision

TABLE 1.7-1(Sh. 26 of 54)
Electrical and Instrumentation and Control Drawings

FSAR Figure No. [(a)]	FSAR Section No.	Drawing Type	Controlled Drawing No.	Controlled Drawing Title	Revision No.	Revision Date

TABLE 1.7-1(Sh. 27 of 54)

Electrical and Instrumentation and Control Drawings

FSAR Figure No. (a)	FSAR Section No.	Drawing Type	Controlled Drawing No.	Controlled Drawing Title	Revision No.	Revision Date

TABLE 1.7-1(Sh. 28 of 54)
Electrical and Instrumentation and Control Drawings

FSAR Figure No. (a)	FSAR Section No.	Drawing Type	Controlled Drawing No.	Controlled Drawing Title	Revision No.	Revision Date

TABLE 1.7-1(Sh. 29 of 54)
Electrical and Instrumentation and Control Drawings

FSAR Figure No. [(a)]	FSAR Section No.	Drawing Type	Controlled Drawing No.	Controlled Drawing Title	Revision No.	Revision Date

TABLE 1.7-1(Sh. 30 of 54)
Electrical and Instrumentation and Control Drawings

FSAR Figure No. [(a)]	FSAR Section No.	Drawing Type	Controlled Drawing No.	Controlled Drawing Title	Revision No.	Revision Date

TABLE 1.7-1(Sh. 31 of 54)
Electrical and Instrumentation and Control Drawings

FSAR Figure No. [a]	FSAR Section No.	Drawing Type	Controlled Drawing No.	Controlled Drawing Title	Revision No.	Revision

TABLE 1.7-1(Sh. 32 of 54)
Electrical and Instrumentation and Control Drawings

FSAR Figure No. [(a)]	FSAR Section No.	Drawing Type	Controlled Drawing No.	Controlled Drawing Title	Revision No.	Revision

TABLE 1.7-1(Sh. 33 of 54)
Electrical and Instrumentation and Control Drawings

FSAR Figure No. [(a)]	FSAR Section No.	Drawing Type	Controlled Drawing No.	Controlled Drawing Title	Revision No.	Revision

TABLE 1.7-1(Sh. 34 of 54)
Electrical and Instrumentation and Control Drawings

FSAR Figure No. [(a)]	FSAR Section No.	Drawing Type	Controlled Drawing No.	Controlled Drawing Title	Revision No.	Revision Date

TABLE 1.7-1(Sh. 35 of 54)
Electrical and Instrumentation and Control Drawings

FSAR Figure No. [a]	FSAR Section No.	Drawing Type	Controlled Drawing No.	Controlled Drawing Title	Revision No.	Revision

TABLE 1.7-1(Sh. 36 of 54)
Electrical and Instrumentation and Control Drawings

FSAR Figure No.[(a)]	FSAR Section No.	Drawing Type	Controlled Drawing No.	Controlled Drawing Title	Revision No.	Revision

TABLE 1.7-1(Sh. 37 of 54)
Electrical and Instrumentation and Control Drawings

FSAR Figure No. (a)	FSAR Section No.	Drawing Type	Controlled Drawing No.	Controlled Drawing Title	Revision No.	Revision Date

TABLE 1.7-1(Sh. 38 of 54)
Electrical and Instrumentation and Control Drawings

FSAR Figure No. [a]	FSAR Section No.	Drawing Type	Controlled Drawing No.	Controlled Drawing Title	Revision No.	Revision

TABLE 1.7-1(Sh. 39 of 54)
Electrical and Instrumentation and Control Drawings

FSAR Figure No. [(a)]	FSAR Section No.	Drawing Type	Controlled Drawing No.	Controlled Drawing Title	Revision No.	Revision

TABLE 1.7-1(Sh. 40 of 54)
Electrical and Instrumentation and Control Drawings

FSAR Figure No. [(a)]	FSAR Section No.	Drawing Type	Controlled Drawing No.	Controlled Drawing Title	Revision No.	Revision

TABLE 1.7-1(Sh. 41 of 54)
Electrical and Instrumentation and Control Drawings

FSAR Figure No. [(a)]	FSAR Section No.	Drawing Type	Controlled Drawing No.	Controlled Drawing Title	Revision No.	Revision Date

TABLE 1.7-1(Sh. 42 of 54)
Electrical and Instrumentation and Control Drawings

FSAR Figure No. [a]	FSAR Section No.	Drawing Type	Controlled Drawing No.	Controlled Drawing Title	Revision No.	Revision

TABLE 1.7-1(Sh. 43 of 54)
Electrical and Instrumentation and Control Drawings

FSAR Figure No. [a]	FSAR Section No.	Drawing Type	Controlled Drawing No.	Controlled Drawing Title	Revision No.	Revision

TABLE 1.7-1(Sh. 44 of 54)
Electrical and Instrumentation and Control Drawings

FSAR Figure No. [(a)]	FSAR Section No.	Drawing Type	Controlled Drawing No.	Controlled Drawing Title	Revision No.	Revision

TABLE 1.7-1(Sh. 45 of 54)
Electrical and Instrumentation and Control Drawings

FSAR Figure No. [(a)]	FSAR Section No.	Drawing Type	Controlled Drawing No.	Controlled Drawing Title	Revision No.	Revision

TABLE 1.7-1(Sh. 46 of 54)
Electrical and Instrumentation and Control Drawings

FSAR Figure No. [(a)]	FSAR Section No.	Drawing Type	Controlled Drawing No.	Controlled Drawing Title	Revision No.	Revision

TABLE 1.7-1(Sh. 47 of 54)
Electrical and Instrumentation and Control Drawings

FSAR Figure No. [(a)]	FSAR Section No.	Drawing Type	Controlled Drawing No.	Controlled Drawing Title	Revision No.	Revision Date

TABLE 1.7-1(Sh. 48 of 54)
Electrical and Instrumentation and Control Drawings

FSAR Figure No.[(a)]	FSAR Section No.	Drawing Type	Controlled Drawing No.	Controlled Drawing Title	Revision No.	Revision Date

TABLE 1.7-1(Sh. 49 of 54)
Electrical and Instrumentation and Control Drawings

FSAR Figure No. [(a)]	FSAR Section No.	Drawing Type	Controlled Drawing No.	Controlled Drawing Title	Revision No.	Revision

TABLE 1.7-1(Sh. 50 of 54)
Electrical and Instrumentation and Control Drawings

FSAR Figure No. [(a)]	FSAR Section No.	Drawing Type	Controlled Drawing No.	Controlled Drawing Title	Revision No.	Revision Date

TABLE 1.7-1(Sh. 51 of 54)
Electrical and Instrumentation and Control Drawings

FSAR Figure No. [(a)]	FSAR Section No.	Drawing Type	Controlled Drawing No.	Controlled Drawing Title	Revision No.	Revision

TABLE 1.7-1(Sh. 52 of 54)
Electrical and Instrumentation and Control Drawings

FSAR Figure No. [(a)]	FSAR Section No.	Drawing Type	Controlled Drawing No.	Controlled Drawing Title	Revision No.	Revision Date

TABLE 1.7-1(Sh. 53 of 54)

Electrical and Instrumentation and Control Drawings

FSAR Figure No. (a)	FSAR Section No.	Drawing Type	Controlled Drawing No.	Controlled Drawing Title	Revision No.	Revision

TABLE 1.7-1(Sh. 54 of 54)
Electrical and Instrumentation and Control Drawings

FSAR Figure No. (a)	FSAR Section No.	Drawing Type	Controlled Drawing No.	Controlled Drawing Title	Revision No.	Revision Date

TABLE 1.7-2(Sh. 1 of 23)
Piping and Instrumentation Diagrams

FSAR Figure No.	FSAR Section No.	Drawing Type	Controlled Drawing No.	Controlled Drawing Title	Revision No.	Revision Date

TABLE 1.7-2(Sh. 2 of 23)
Piping and Instrumentation Diagrams

FSAR Figure No.	FSAR Section No.	Drawing Type	Controlled Drawing No.	Controlled Drawing Title	Revision No.	Revision Date

TABLE 1.7-2(Sh. 3 of 23)
Piping and Instrumentation Diagrams

FSAR Figure No.	FSAR Section No.	Drawing Type	Controlled Drawing No.	Controlled Drawing Title	Revision No.	Revision Date

TABLE 1.7-2(Sh. 4 of 23)
Piping and Instrumentation Diagrams

FSAR Figure No.	FSAR Section No.	Drawing Type	Controlled Drawing No.	Controlled Drawing Title	Revision No.	Revision Date

TABLE 1.7-2(Sh. 5 of 23)
Piping and Instrumentation Diagrams

FSAR Figure No.	FSAR Section No.	Drawing Type	Controlled Drawing No.	Controlled Drawing Title	Revision No.	Revision Date

TABLE 1.7-2(Sh. 6 of 23)
Piping and Instrumentation Diagrams

FSAR Figure No.	FSAR Section No.	Drawing Type	Controlled Drawing No.	Controlled Drawing Title	Revision No.	Revision Date

TABLE 1.7-2(Sh. 7 of 23)
Piping and Instrumentation Diagrams

FSAR Figure No.	FSAR Section No.	Drawing Type	Controlled Drawing No.	Controlled Drawing Title	Revision No.	Revision Date

TABLE 1.7-2(Sh. 8 of 23)
Piping and Instrumentation Diagrams

FSAR Figure No.	FSAR Section No.	Drawing Type	Controlled Drawing No.	Controlled Drawing Title	Revision No.	Revision Date

TABLE 1.7-2(Sh. 9 of 23)
Piping and Instrumentation Diagrams

FSAR Figure No.	FSAR Section No.	Drawing Type	Controlled Drawing No.	Controlled Drawing Title	Revision No.	Revision Date

TABLE 1.7-2(Sh. 10 of 23)
Piping and Instrumentation Diagrams

FSAR Figure No.	FSAR Section No.	Drawing Type	Controlled Drawing No.	Controlled Drawing Title	Revision No.	Revision Date

TABLE 1.7-2(Sh. 11 of 23)
Piping and Instrumentation Diagrams

FSAR Figure No.	FSAR Section No.	Drawing Type	Controlled Drawing No.	Controlled Drawing Title	Revision No.	Revision Date

TABLE 1.7-2(Sh. 12 of 23)
Piping and Instrumentation Diagrams

FSAR Figure No	FSAR Section No	Drawing Type	Controlled Drawing No	Controlled Drawing Title	Revision No	Revision Date

TABLE 1.7-2(Sh. 13 of 23)
Piping and Instrumentation Diagrams

FSAR Figure No.	FSAR Section No.	Drawing Type	Controlled Drawing No.	Controlled Drawing Title	Revision No.	Revision Date

TABLE 1.7-2(Sh. 14 of 23)
Piping and Instrumentation Diagrams

FSAR Figure No.	FSAR Section No.	Drawing Type	Controlled Drawing No.	Controlled Drawing Title	Revision No.	Revision Date

TABLE 1.7-2(Sh. 15 of 23)
Piping and Instrumentation Diagrams

FSAR Figure No.	FSAR Section No.	Drawing Type	Controlled Drawing No.	Controlled Drawing Title	Revision No.	Revision Date

TABLE 1.7-2(Sh. 16 of 23)
Piping and Instrumentation Diagrams

FSAR Figure No.	FSAR Section No.	Drawing Type	Controlled Drawing No.	Controlled Drawing Title	Revision No.	Revision Date

TABLE 1.7-2(Sh. 17 of 23)
Piping and Instrumentation Diagrams

FSAR Figure No.	FSAR Section No.	Drawing Type	Controlled Drawing No.	Controlled Drawing Title	Revision No.	Revision Date

TABLE 1.7-2(Sh. 18 of 23)
Piping and Instrumentation Diagrams

FSAR Figure No.	FSAR Section No.	Drawing Type	Controlled Drawing No.	Controlled Drawing Title	Revision No.	Revision Date

TABLE 1.7-2(Sh. 19 of 23)
Piping and Instrumentation Diagrams

FSAR Figure No.	FSAR Section No.	Drawing Type	Controlled Drawing No.	Controlled Drawing Title	Revision No.	Revision Date

TABLE 1.7-2(Sh. 20 of 23)
Piping and Instrumentation Diagrams

FSAR Figure No.	FSAR Section No.	Drawing Type	Controlled Drawing No.	Controlled Drawing Title	Revision No.	Revision Date

TABLE 1.7-2(Sh. 21 of 23)
Piping and Instrumentation Diagrams

FSAR Figure No.	FSAR Section No.	Drawing Type	Controlled Drawing No.	Controlled Drawing Title	Revision No.	Revision Date

TABLE 1.7-2(Sh. 22 of 23)
Piping and Instrumentation Diagrams

FSAR Figure No.	FSAR Section No.	Drawing Type	Controlled Drawing No.	Controlled Drawing Title	Revision No.	Revision Date

TABLE 1.7-2(Sh. 23 of 23)
Piping and Instrumentation Diagrams

FSAR Figure No.	FSAR Section No.	Drawing Type	Controlled Drawing No.	Controlled Drawing Title	Revision No.	Revision Date

TABLE 1.7-3(Sh. 1 of 18)
Other Diagrams

FSAR Figure No.	FSAR Section No.	Drawing Type	Controlled Drawing No.	Controlled Drawing Title	Revision No.	Revision Date

TABLE 1.7-3(Sh. 2 of 18)
Other Diagrams

FSAR Figure No.	FSAR Section No.	Drawing Type	Controlled Drawing No.	Controlled Drawing Title	Revision No.	Revision Date

TABLE 1.7-3(Sh. 3 of 18)
Other Diagrams

FSAR Figure No.	FSAR Section No.	Drawing Type	Controlled Drawing No.	Controlled Drawing Title	Revision No.	Revision Date

TABLE 1.7-3(Sh. 4 of 18)
Other Diagrams

FSAR Figure No.	FSAR Section No.	Drawing Type	Controlled Drawing No.	Controlled Drawing Title	Revision No.	Revision Date

TABLE 1.7-3(Sh. 5 of 18)
Other Diagrams

FSAR Figure No.	FSAR Section No.	Drawing Type	Controlled Drawing No.	Controlled Drawing Title	Revision No.	Revision Date

TABLE 1.7-3(Sh. 6 of 18)
Other Diagrams

FSAR Figure No.	FSAR Section No.	Drawing Type	Controlled Drawing No.	Controlled Drawing Title	Revision No.	Revision Date

TABLE 1.7-3(Sh. 7 of 18)
Other Diagrams

FSAR Figure No.	FSAR Section No.	Drawing Type	Controlled Drawing No.	Controlled Drawing Title	Revision No.	Revision Date

TABLE 1.7-3(Sh. 8 of 18)
Other Diagrams

FSAR Figure No.	FSAR Section No.	Drawing Type	Controlled Drawing No.	Controlled Drawing Title	Revision No.	Revision Date

TABLE 1.7-3(Sh. 9 of 18)
Other Diagrams

FSAR Figure No.	FSAR Section No.	Drawing Type	Controlled Drawing No.	Controlled Drawing Title	Revision No.	Revision Date

TABLE 1.7-3(Sh. 10 of 18)
Other Diagrams

FSAR Figure No.	FSAR Section No.	Drawing Type	Controlled Drawing No.	Controlled Drawing Title	Revision No.	Revision Date

TABLE 1.7-3(Sh. 11 of 18)
Other Diagrams

FSAR Figure No.	FSAR Section No.	Drawing Type	Controlled Drawing No.	Controlled Drawing Title	Revision No.	Revision Date

TABLE 1.7-3(Sh. 12 of 18)
Other Diagrams

FSAR Figure No.	FSAR Section No.	Drawing Type	Controlled Drawing No.	Controlled Drawing Title	Revision No.	Revision Date

TABLE 1.7-3(Sh. 13 of 18)
Other Diagrams

FSAR Figure No.	FSAR Section No.	Drawing Type	Controlled Drawing No.	Controlled Drawing Title	Revision No.	Revision Date

TABLE 1.7-3(Sh. 14 of 18)
Other Diagrams

FSAR Figure No.	FSAR Section No.	Drawing Type	Controlled Drawing No.	Controlled Drawing Title	Revision No.	Revision Date

TABLE 1.7-3(Sh. 15 of 18)
Other Diagrams

FSAR Figure No.	FSAR Section No.	Drawing Type	Controlled Drawing No.	Controlled Drawing Title	Revision No.	Revision Date

TABLE 1.7-3(Sh. 16 of 18)
Other Diagrams

FSAR Figure No.	FSAR Section No.	Drawing Type	Controlled Drawing No.	Controlled Drawing Title	Revision No.	Revision Date

TABLE 1.7-3(Sh. 17 of 18)
Other Diagrams

FSAR Figure No.	FSAR Section No.	Drawing Type	Controlled Drawing No.	Controlled Drawing Title	Revision No.	Revision Date

TABLE 1.7-3(Sh. 18 of 18)
Other Diagrams

FSAR Figure No.	FSAR Section No.	Drawing Type	Controlled Drawing No.	Controlled Drawing Title	Revision No.	Revision Date

1.8 COMPLIANCE WITH U.S.NRC REGULATORY GUIDANCE

1.8.1 Regulatory Guides

Table 1.8-1 lists the Regulatory Guides that are addressed in the YGN 3&4 design. The applicable Regulatory Guide version and the FSAR section in which each Regulatory Guide is addressed are noted.

Regulatory Guides listed in Table 1.8-1 include all guides issued prior to and in effect as of December 31, 1985. Regulatory Guides issued after this date are not applicable to YGN 3&4, unless specifically based upon a significant safety concern or Owner preference.

Appendix 1A contains the YGN 3&4 positions with respect to the Regulatory Guides.

1.8.2 TMI Action Items

USNRC action items resulting from the TMI-2 accident that are applicable to YGN 3&4 are discussed in Appendix 1B.

TABLE 1.8-1 (Sh. 1 of 21)

REGULATORY GUIDES

Document/Title	Original or Revision Issue Date	Reference FSAR Section
Reg. Guide 1.1 - Net Positive Suction Head for Emergency Core Cooling and Containment Heat Removal System Pumps	11/70	6.3
Reg. Guide 1.2 - Thermal Shock to Reactor Pressure Vessels	11/70	5.2.3
Reg. Guide 1.3 -		Not Applicable (BWR)
Reg. Guide 1.4 - Assumptions Used for Evaluating the Potential Radiological Consequences of a LOCA for Pressurized Water Reactors	Revision 2 6/74	2.3.4, 15.6.5, Appendix 15E
Reg. Guide 1.5 -		Not Applicable (BWR)
Reg. Guide 1.6 - Independence between Redundant Standby (Onsite) Power Sources and Between Their Distribution Systems	3/71	8.3.1.1, 8.3.1.2, 8.3.2.1, 8.3.2.2
Reg. Guide 1.7 - Control of Combustible Gas Concentrations in Containment following LOCA	Revision 2 11/78	6.2.5
Reg. Guide 1.8 - Personnel Selection and Training	Revision 1-R 5/77	13.1, 13.2
Reg. Guide 1.9 - Selection, Design, and Qualification of Diesel-Generator Units Used as Standby (Onsite) Electric Power Systems at Nuclear Power Plants	Revision 2 12/79	8.3.1

TABLE 1.8-1 (Sh. 1 of 21)

REGULATORY GUIDES

Document/Title	Original or Revision Issue Date	Reference FSAR Section
Reg. Guide 1.9 - Selection, Design, Qualification, and Testing of Emergency Diesel Generator Units Used as Class 1E Onsite Electric Power System at Nuclear Power Plants	Revision 3 7/93	8.3.1 ~~Chapter 16~~ 삭제

TABLE 1.8-1 (Sh. 2 of 21)

Document/Title	Original or Revision Issue Date	Reference FSAR Section	
Reg. Guide 1.10 -	Withdrawn	NA	
Reg. Guide 1.11- Instrument Lines Penetrating Primary Reactor Containment	3/71 Supplement 2/72	7.1.2	
Reg. Guide 1.12- Instrumentation for Earthquakes	Withdrawn	NA	802
Reg. Guide 1.13- Spent-Fuel Storage Facility Design Basis	Revision 1 12/75	9.1.2, 9.1.3	
Reg. Guide 1.14 - Reactor Coolant Pump Flywheel Integrity	Revision 1 8/75	5.4.1	
Reg. Guide 1.15 -	Withdrawn	NA	
Reg. Guide 1.16 - Reporting of Operating Information	Revision 4 8/75	~~16.6~~ ITS chapter3 5.0	
Reg. Guide 1.17 - Protection of Nuclear Power Plants Against Industrial Sabotage	Revision 1 6/73	13.6	
Reg. Guide 1.18 -	Withdrawn	NA	
Reg. Guide 1.19 -	Withdrawn	NA	
Reg. Guide 1.20 - Comprehensive Vibration Assessment Program for Reactor Internals During Preoperational and Initial Startup Testing	Revision 2 5/76	3.9.2	

TABLE 1.8-1 (Sh. 3 of 21)

Document/Title	Original or Revision Issue Date	Reference FSAR Section
Reg. Guide 1.21 - Measuring, Evaluating, and Reporting Radioactivity in Solid Wastes and Releases of Radioactive Materials in Liquid and Gaseous Effluents from Light-Water-Cooled Nuclear Power Plants	Revision 1 6/74	11.5, ~~16.6~~ ITS chapter 3 5.0
Reg. Guide 1.22 - Periodic Testing of Protection Systems Actuation Functions	Revision 0 2/72	7.1.2, 7.3.1,
Reg. Guide 1.23 - Onsite Meteorological Monitoring Programs	2/72	2.3
Reg. Guide 1.24 - Assumptions Used for Evaluating the Potential Radiological Consequences of a Pressurized Water Reactor Radioactive Gas Storage Tank Failure	3/72	15.7.1
Reg. Guide 1.25 - Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel-Handling Accident in the Fuel Handling and Storage Facility For Boiling and Pressurized Water Reactors	3/72	15.7.4
Reg. Guide 1.26 - Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants	Revision 3 2/76 (Draft)	3.2.2
Reg. Guide 1.27 - Ultimate Heat Sink for Nuclear Power Plants	Revision 2 1/76	9.2.5

TABLE 1.8-1(Sh.4 of 21)

Document/Title	Original or Revision Issue Date	Reference FSAR Section
Reg. Guide 1.28 - Quality Assurance Program Requirements (Design and Construction)	Revision 3 8/85	17.1, 17.2
Reg. Guide 1.29 - Seismic Design Classification	Revision 3 9/78	3.2.1, 3.10
Reg. Guide 1.30 - Quality Assurance Requirements for the Installation, Inspection and Testing of Instrumentation and Electrical Equipment	8/72	No specific section.
Reg. Guide 1.31* - Control of Ferrite Content in Stainless Steel Weld Metal	Revision 3 4/78	5.2.3
Reg. Guide 1.32 - Criteria for Safety-Related Electric Power Systems for Nuclear Power Plants	Revision 2 2/77	8.1.2., 8.1.3, 8.3.1.1, 8.3.1.2, 8.3.2.2
Reg. Guide 1.33 - Quality Assurance Program Requirements (Operation)	Revision 2 12/72	17.2
Reg. Guide 1.34 - Control of Electroslag Weld Properties	12/72	5.2.3.3.2.2
Reg. Guide 1.35 - Inservice Inspection of Ungrouted Tendons in Prestressed Concrete Containments	Revision 3 7/90	3.8.1, ~~16.4.6~~ ITS chapter1 3.6
Reg. Guide 1.35.1 - Determining Prestressing Forces for Inspection of Prestressed Concrete Containments	Revision 0 7/90	3.8.1, ~~16.4.6~~

※ : For replacement steam generators, Revision 4 (October 2013) of the regulatory guide is applied

TABLE 1.8-1 (Sh. 5 of 21)

Document/Title	Original or Revision Issue Date	Reference FSAR Section
Reg. Guide 1.36 - Nonmetallic Thermal Insulation for Austenitic Stainless Steel	2/73	5.2.3
Reg. Guide 1.37 - Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants	3/73	17.1, 17.2
Reg. Guide 1.38 - Quality Assurance Requirements for Packaging, Shipping, Receiving, Storage, and Handling of Items for Water - Cooled Nuclear Power Plants	Revision 2 5/77	17.1, 17.2
Reg. Guide 1.39 - Housekeeping Requirements for Water-Cooled Nuclear Power Plants	Revision 2 9/77	17.1, 17.2
Reg. Guide 1.40 - Qualification Tests of Continuous-Duty Motors Installed Inside of Containment of Water-Cooled Nuclear Power Plants	Revision 0 3/73	3.11
Reg. Guide 1.41 - Preoperational Testing of Redundant Onsite Electric Power Systems to Verify Proper Load Group Assignments	Revision 0 3/73	8.3.1.1, 8.3.2.2
Reg. Guide 1.42 -	Withdrawn	NA

TABLE 1.8-1 (Sh.6 of 21)

Document/Title	Original or Revision Issue Date	Reference FSAR Section
Reg. Guide 1.43* - Control of Stainless Steel Weld Cladding of Low-Alloy Steel Components	5/73	5.2.3
Reg. Guide 1.44** - Control of the Use of Sensitized Stainless Steel	5/73	5.2.3
Reg. Guide 1.45 - Reactor Coolant Pressure Boundary Leakage Detection Systems	Revision 0 5/73	5.2.5
Reg. Guide 1.46 -	Withdrawn	NA
Reg. Guide 1.47 - Bypassed and Inoperable Status Indication for Nuclear Power Plant Safely Systems	Revision 0 5/73	7.1.2, 7.5
Reg. Guide 1.48 -	Withdrawn	NA
Reg. Guide 1.49 - Power Levels of Water-Cooled Nuclear Power Plants	Revision 1 12/73	6.3 Chapter 15.0
Reg. Guide 1.50*** - Control of Preheat Temperature for Welding of Low-Alloy Steel	5/73	5.2.3
Reg. Guide 1.51 -	Withdrawn	NA
Reg. Guide 1.52 - Design, Testing, and Maintenance Criteria for Engineered-Safety-Feature Atmosphere Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants	Revision 2 3/78	6.5, 9.4, 12.3

* : For replacement steam generators, Revision 1 (March 2011) of the regulatory guide 1.43 is applied

** : For replacement steam generators, Revision 1 (March 2011) of the regulatory guide 1.44 is applied

*** : For replacement steam generators, Revision 1 (March 2011) of the regulatory guide 1.50 is applied

TABLE 1.8-1 (Sh. 7 of 21)

Document/Title	Original or Revision Issue Date	Reference FSAR Section
Reg. Guide 1.53 - Application of the Single-Failure Criterion to Nuclear Power Plant Protection Systems	Revision 0 6/73	7.1.2
Reg. Guide 1.54 - Quality Assurance Requirements for Protective Coatings Applied to Water-Cooled Nuclear Power Plants	Revision 0 6/73	6.1.2
Reg. Guide 1.55 -	Withdrawn	NA
Reg. Guide 1.56 -		NA (BWR)
Reg. Guide 1.57 - Design Limits and Loading Combinations for Metal Primary Reactor Containment System Components	Revision 0 6/73	3.8.2
Reg. Guide 1.58 - Qualification of Nuclear Power Plant Inspection, Examination and Testing Personnel	Revision 1 9/80	NA
Reg. Guide 1.59 - Design Basis Floods for Nuclear Power Plants	Revision 2 8/77	2.4
Reg. Guide 1.60 - Design Response Spectra for Seismic Design of Nuclear Power Plants	Revision 1 12/73	2.5
Reg. Guide 1.61 - Damping Values for Seismic Design of Nuclear Power Plants	Revision 0 10/73	3.7.1, 3.7.2

TABLE 1.8-1 (Sh. 8 of 21)

Document/Title	Original or Revision Issue Date	Reference FSAR Section
Reg. Guide 1.62 - Manual Initiation of Protective Actions	Revision 0 10/73	7.1.2
Reg. Guide 1.63 - Electric Penetration Assemblies in Containment Structures for Light-Water-Cooled Nuclear Power Plants	Revision 3 2/87	7.1.2
Reg. Guide 1.64 - Quality Assurance Requirements for the Design of Nuclear Power Plants	Revision 2 6/76	NA
Reg. Guide 1.65 - Material and Inspections for Reactor Vessel Closure Studs	10/73	5.3.1
Reg. Guide 1.66 -	Withdrawn	NA
Reg. Guide 1.67 -	Withdrawn	NA
Reg. Guide 1.68 - Initial Test Programs for Water-Cooled Reactor Power Plants	Revision 2 8/78	14.2.7
Reg. Guide 1.68.1 -		NA (BWR)
Reg. Guide 1.68.2 - Initial Startup Test to Demonstrate Remote Shutdown Capability for Water-Cooled Nuclear Power Plants	Revision 1 7/78	14.2
Reg. Guide 1.68.3 - Preoperational Testing of Instrument and Control Air Systems	Revision 0 4/82	14.2

TABLE 1.8-1 (Sh.9 of 21)

Document/Title	Original or Revision Issue Date	Reference FSAR Section
Reg. Guide 1.69 - Concrete Radiation Shields for Nuclear Power Plants	12/73	12.3
Reg. Guide 1.70 - Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants	Revision 3 11/78	All
Reg. Guide 1.71* - Welder Qualification for Areas of Limited Accessibility	12/73	5.2.3, 5.3.1,
Reg. Guide 1.72 -		NA (spray pond not used)
Reg. Guide 1.73 - Qualification Tests of Electric Valve Operators Installed Inside the Containment of Nuclear Power Plants	Revision 0 1/74	7.1.2, 3.11
Reg. Guide 1.74 - Quality Assurance Terms and Definitions	Revision 0 1/74	NA
Reg. Guide 1.75 - Physical Independence of Electric Systems	Revision 2 9/78	No specific section.
Reg. Guide 1.76 - Design Bases Torando for Nuclear Power Plants	Revision 0 4/74	2.3, 3.3
Reg. Guide 1.77 - Assumptions Used for Evaluating a Control Rod Ejextion Accident for Pressurized Water Reactors	5/74	15.4.8

* : For replacement steam generators, Revision 1 (March 2011) of the regulatory guide 1.43 is applied

TABLE 1.8-1 (Sh.10 of 21)

Document/Title	Original or Revision Issue Date	Reference FSAR Section
Reg. Guide 1.78 - Assumptions for Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release	6/74	2.2, 6.4
Reg. Guide 1.79 - Preoperational Testing of Emergency Core Cooling Systems for Pressurized Water Systems	Revision 1 9/75	3.1.2.33, 14.3
Reg. Guide 1.80 -		Withdrawn
Reg Guide 1.81 - Shared Emergency and Shutdown Electric Systems for Multi-Unit Nuclear Power Plants	Revision 1 1/75	8.3.1
Reg. Guide 1.82 - Water Sources for Long-Term Recirculation Cooling Following a Loss-of-Coolant Accident	Revision 3 11/03	6.2.2
Reg. Guide 1.83 - Inservice Inspection of Pressurized Water Reactor Steam Generator Tubes	Revision 1 7/75	5.2.4, ~~Chapter 16.0~~ ITS
Reg. Guide 1.84* - Design and Fabrication Code Case Acceptability ASME Section III, Division 1	Revision 23 9/85	5.2
Reg. Guide 1.85* - Materials Code Acceptability ASME Section III, Division 1	Revision 23 9/85	5.2
Reg. Guide 1.86 - Termination of Operating Licenses for Nuclear Reactors	6/74	No specific section

* : For replacement steam generators, Revision 35 (October 2010) of the regulatory guide is applied.
In case of inconel 690(I-52 or I-52M) welding, ASME Code Case 2142-2 is used.

** : For replacement steam generators, this regulatory guide incorporated into the regulatory guide 1.84

TABLE 1.8-1 (Sh. 11 of 21)

Document/Title	Original or Revision Issue Date	Reference FSAR Section
Reg. Guide 1.87 - Guidance for Construction of Class 1 Components in Elevated-Temperature Reactors	Revision 1 6/75	NA (ETR)
Reg. Guide 1.88 - Collection, Storage, and Maintenance of Nuclear Power Plant Quality Assurance Records	Revision 2 10/76	NA
Reg. Guide 1.89 - Environmental Qualification of Certain Electrical Equipment Important to Safety for Nuclear Power Plants	Revision 1 6/84	3.11
Reg. Guide 1.90 - Inservice Inspection of Prestressed Concrete Containment Structures with Grouted Tendons	Revision 1 8/77	NA (grouted tendons not used)
Reg. Guide 1.91 - Evaluation of Explosions Postulated to Occur on Transportation Routes Near Nuclear Power Plants	Revision 1 2/78	2.2
Reg. Guide 1.92 - Combining Modal Responses and Spatial Components in Seismic Response Analysis	Revision 1 2/76	3.7.2, 3.7.3
Reg. Guide 1.93 - Availability of Electric Power Sources	Revision 0 12/74	~~16.3/4.8~~ ITS Chapter 1 3.8
Reg. Guide 1.94 - Quality Assurance Requirements for Installation, Inspection, and Testing of Structural Concrete and Structural Steel During the Construction Phase of Nuclear Power Plants	Revision 1 4/76	17.1, 17.2

TABLE 1.8-1 (Sh. 12 of 21)

Document/Title	Original or Revision Issue Date	Reference FSAR Section
Reg. Guide 1.95 - Protection of Nuclear Power Plant Control Room Operators Against an Accidental Chlorine Release	Revision 1 1/77	2.2, 6.4
Reg. Guide 1.96 -		NA (BWR)
Reg. Guide 1.97 - Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident	Revision 3 5/83	3.1, 7.5
Reg. Guide 1.98 -		NA (BWR)
Reg. Guide 1.99 - Radiation Embrittlement of Reactor Vessel Materials	Revision 2 5/88	5.3
Reg. Guide 1.100 - Seismic Qualification of Electrical Equipment for Nuclear Power Plants	Revision 1 8/77	3.10, 7.1.2
Reg. Guide 1.101 - Emergency Planning and Preparedness for Nuclear Power Reactors	Revision 2 10/81	13.3
Reg. Guide 1.102 - Flood Protection for Nuclear Power Plants	Revision 1 9/76	2.4, 3.4
Reg. Guide 1.103 -	Withdrawn	NA
Reg. Guide 1.104 -	Withdrawn	NA
Reg. Guide 1.105 - Instrument Setpoints	Revision 2 2/86	7.1.2

TABLE 1.8-1 (Sh. 13 of 21)

Document/Title	Original or Revision Issue Date	Reference FSAR Section
Reg. Guide 1.106 - Thermal Overload Protection for Electric Motors on Motor-Operated Valves	Revision 1 3/77	~~Chapter 16~~ 13.7
Reg. Guide 1.107 -	---	NA (grouted tendons not used)
Reg. Guide 1.108 - Periodic Testing of Diesel Generators Used As Onsite Electric Power Systems at Nuclear Power Plants	Revision 1 8/77	8.3.1
Reg. Guide 1.109 - Calculation of Annual Doses to Man From Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I	Revision 1 10/77	11.2.3, 11.3.3, ~~Chapter 16.0~~ ITS
Reg. Guide 1.110 - Cost-Benefit Analysis for Radwaste Systems for Light-Water-Cooled Nuclear Power Reactors	Revision 0 3/76	11.2, 11.3
Reg. Guide 1.111 - Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water-Cooled Reactors	Revision 1 7/77	2.3
Reg. Guide 1.112 - Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Light-Water-Cooled Power Reactors	Revision 0-R 5/77	11.2, 11.3

TABLE 1.8-1 (Sh. 14 of 21)

Document/Title	Original or Revision Issue Date	Reference FSAR Section
Reg. Guide 1.113 - Estimating Aquatic Dispersion of Effluents from Accidental and Routine Reactor Releases for the Purpose of Implementing Appendix I	Revision 1 4/77	11.2
Reg. Guide 1.114 - Guidance on Being Operator at the Controls of a Nuclear Power Plant	Revision 1 11/76	13.1.3, ~~16.6~~ ITS chapter 3
Reg. Guide 1.115 - Protection Against Low-Trajectory Turbine Missiles	Revision 1 7/77	3.5
Reg. Guide 1.116 - Quality Assurance Requirements for Installation, Inspection, and Testing of Mechanical Equipment and Systems	Revision 0-R 5/77	17.1, 17.2
Reg. Guide 1.117 - Tornado Design Classification	Revision 1 4/78	3.3
Reg. Guide 1.118 - Periodic Testing of Electric Power and Protection Systems	Revision 2 16/78	7.1.2
Reg. Guide 1.119 -	Withdrawn	NA
Reg. Guide 1.120 - Fire Protection Guidelines for Nuclear Power Plants	Draft Revision 1 11/77	Draft Reg. Guide NA
Reg. Guide 1.121 - Bases for Plugging Degraded PWR Steam Generator Tubes	8/76	~~Chapter 16~~ ITS

TABLE 1.8-1 (Sh. 15 of 21)

Document/Title	Original or Revision Issue Date	Reference FSAR Section
Reg. Guide 1.122 - Development of Floor Design Response Spectra for Seismic Design of Floor-Supported Equipment or Components	Revision 1 2/78	3.7.2
Reg. Guide 1.123 - Quality Assurance Requirements for Control of Procurement of Items and Services for Nuclear Power Plants	Revision 1 7/77	NA
Reg. Guide 1.124 - Service Limits and Loading Combinations for Class-1 Linear-Type Component Supports	Revision 1 1/78	3.9.3
Reg. Guide 1.125 - Physical Models for Design and Operation of Hydraulic Structures and Systems for Nuclear Power Plants	Revision 1 10/78	NA No physical model
Reg. Guide 1.126 - An Acceptable Model and Related Statistical Methods for the Analysis of Fuel Densification	Revision 1 3/78	4.2.1
Reg. Guide 1.127 - Inspection of Water-Control Structures Associated with Nuclear Power Plants	Revision 1 3/78	NA Water-Control Structures not Used
Reg. Guide 1.128 - Installation Design and Installation of Large Lead Storage Batteries for Nuclear Power Plants	Revision 1 10/78	No Specific Section

TABLE 1.8-1 (Sh. 16 of 21)

Document/Title	Original or Revision Issue Date	Reference FSAR Section
Reg. Guide 1.129 - Maintenance, Testing, and Replacement of Large Lead Storage Batteries for Nuclear Power Plants	Revision 1 2/78	~~16.3/4.8.4~~ ITS chapter 1 3.8
Reg. Guide 1.130 - Service Limits and Loading Combinations for Class 1 Plate-and-Shell-Type Component Supports	Revision 1 10/78	3.9.3
Reg. Guide 1.131 - Qualification Tests of Electric Cables, Field Splices, and Connections for Light-Water-Cooled Nuclear Power Plants	Revision 0 8/77	3.11, 6.2
Reg. Guide 1.132 - Site Investigations for Foundations of Nuclear Power Plants	Revision 1 3/79	2.5
Reg. Guide 1.133 - Loose-Part Detection Program for the Primary System of Light-Water-Cooled Reactors	Revision 1 5/81	7.1.2
Reg. Guide 1.134 - Medical Evaluation of Nuclear Power Plant Personnel Requiring Operator Licenses	Revision 1 3/79	No Specific Section
Reg. Guide 1.135 - Normal Water Level and Discharge at Nuclear Power Plants	Revision 0 9/77	2.4
Reg. Guide 1.136 - Materials, Construction, and Testing of Concrete Containments	Revision 2 6/81	3.8.1

TABLE 1.8-1 (Sh. 17 of 21)

Document/Title	Original or Revision Issue Date	Reference FSAR Section
Reg. Guide 1.137 - Fuel-Oil Systems for Standby Diesel Generators	Revision 1 10/79	3.2, 9.5
Reg. Guide 1.138 - Laboratory Investigations of Soils for Engineering Analysis and Design of Nuclear Power Plants	Revision 0 4/78	2.5
Reg. Guide 1.139 - Guidance for Residual Heat Removal	5/78	Draft Reg. Guide NA
Reg. Guide 1.140 - Design, Testing, and Maintenance Criteria for Normal Ventilation Exhaust System Air Filtration Adsorption Units of Light-Water-Cooled Nuclear Power Plants	Revision 1 10/79	9.4
Reg. Guide 1.141 - Containment Isolation Provisions for Fluid Systems	Revision 0 4/78	6.2.4
Reg. Guide 1.142 - Safety-Related Concrete Structures for Nuclear Power Plants (Other Than Reactor Vessels and Containments)	Revision 1 10/81	3.8.4
Reg. Guide 1.143 - Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water-Cooled Nuclear Power Plants	Revision 1 10/79	3.2, 11.2, 11.3, 11.4

TABLE 1.8-1 (Sh. 18 of 21)

Document/Title	Original or Revision Issue Date	Reference FSAR Section
Reg. Guide 1.144 - Auditing of Quality Assurance Programs for Nuclear Power Plants	Revision 1 9/80	NA
Reg. Guide 1.145 - Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants	Revision 1 11/82	2.3.4
Reg. Guide 1.146 - Qualification of Quality Assurance Program Audit Personnel for Nuclear Power Plants	Revision 0 8/80	NA
Reg. Guide 1.147*- Inservice Inspection Code Case Acceptability ASME Section XI, Division 1	Revision 4 9/85	5.2.4, 6.6
Reg. Guide 1.148 - Functional Specification for Active Valve Assemblies in Systems Important to Safety in Nuclear Power Plants	3/18	3.9.3, 3.11
Reg. Guide 1.149 - Nuclear Power Plant Simulators for Use in Operator Training	Revision 1 4/87	Appendix 1B, 13.2
Reg. Guide 1.150 - Ultrasonic Testing of Reactor Vessel Welds During Preservice and Inservice Examinations	Revision 1 2/83	5.3.1
Reg. Guide 1.151 - Instrument Sensing Lines	7/83	No specific section.

* : For replacement steam generators, Revision 16 (October 2010) of the regulatory guide is applied

TABLE 1.8-1 (Sh. 19 of 21)

Document/Title	Original or Revision Issue Date	Reference FSAR Section
Reg. Guide 1.152 - Criteria for Programmable Digital Computer System Software in Safety-Related Systems of Nuclear Power Plants	11/85	7.1.2
Reg. Guide 1.153 - Criteria for Power, Instrumentation, and Control Portions of Safety Systems	Revision 0 12/85	7.1.2
Reg. Guide 1.154 - Format and Content of Plant-Specific Pressurized Thermal Shock Safety Analysis Reports	1/87	NA (See Reg. Guide 1.2)
Reg. Guide 1.155 Station Blackout	Revision 1 8/88	3.1.2.5, 9.5.12, 8.3.1
Reg. Guide 1.156 - Environmental Qualification of Connection Assemblies for Nuclear power plants	11/87	NA
Reg. Guide 1.157 - Best-Estimate Calculation of Emergency Core Cooling System Performance	5/89	NA Appendix K Analyses Performed
Reg. Guide 8.2 - Guide for Administrative Practices in Radiation Monitoring	Revision 0 2/73	12.1.1, 12.3.4, 12.5, 13.2
Reg. Guide 8.3 - Film Badge Performance Criteria	Revision 0 2/73	12.5.2, 12.5.3
Reg. Guide 8.4 - Direct-Reading and Indirect-Reading Pocket Dosimeters	Revision 0 2/73	12.5.3

TABLE 1.8-1 (Sh. 20 of 21)

Document/Title	Original or Revision Issue Date	Reference FSAR Section
Reg. Guide 8.5 - Criticality and Other Interior Evacuation Signals	Revision 1 3/81	12.5.3
Reg. Guide 8.7 - Occupational Radiation Exposure Records System	Revision 0 5/73	12.5
Reg. Guide 8.8 - Information Relevant to Ensuring that Occupational Radiation Exposures at Nuclear Power Stations will be as Low as is Reasonably Achievable	Revision 3 6/78	12.1, 12.3, 12.5
Reg. Guide 8.9 - Acceptable Concepts, Models, Equations, and Assumptions for a Bioassy Program	Revision 0 9/73	12.5
Reg. Guide 8.10 - Operating Philosophy for Maintaining Occupational Radiation Exposures as Low as is Reasonably Achievable	Revision 1-R 5/77	12.3.1, 12.5, 13.2
Reg. Guide 8.12 - Criticality Accident Alarm Systems	Revision 1 1/81	9.1
Reg. Guide 8.13 - Instruction Concerning Prenatal Radiation Exposure	Revision 1 11/75	No Specific Section
Reg. Guide 8.14 - Personnel Neutron Dosimeters	Revision 1 8/77	12.5
Reg. Guide 8.15 - Acceptable Programs for Respiratory Protection	Revision 0 10/76	12.5

TABLE 1.8-1 (Sh. 21 of 21)

Document/Title	Original or Revision Issue Date	Reference FSAR Section
Reg. Guide 8.19 - Occupational Radiation Dose Assessment in Light-Water Reactor Power Plants Design Stage Man-Rem Estimates	Revision 1 6/79	12.4

1.9 Improvement Action Items Post Fukushima Daiichi Accident

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

APPENDIX 1A

USNRC REGULATORY GUIDES

REGULATORY GUIDE 1.1

Revision 0, November 1970

NET POSITIVE SUCTION HEAD FOR EMERGENCY CORE COOLING AND CONTAINMENT HEAT REMOVAL SYSTEM PUMPS

The YGN 3&4 complies with the regulatory position with the following exception:

> Calculations of available NPSH for the emergency core cooling and containment heat removal pumps were performed assuming that the containment pressure during postaccident conditions is equal to the vapor pressure of the liquid in the containment. This assumption ensures that the actual available NPSH is always greater than the calculated available NPSH, which meets the intent of the regulatory position.

Compliance with the requirements of this Regulatory Guide is described further in Subsection 6.3.2.2.

REGULATORY GUIDE 1.2

Revision 0, November 1970

THERMAL SHOCK TO REACTOR PRESSURE VESSELS

YGN 3&4 complies with the intent of this Regulatory Guide.

Compliance with the guidance of this regulatory position is described in Subsection 5.2.3.3.1.

REGULATORY GUIDE 1.4

Revision 2, June 1974

ASSUMPTIONS USED FOR EVALUATING THE POTENTIAL RADIOLOGICAL CONSEQUENCES OF A LOSS-OF-COOLANT ACCIDENTS FOR PWRs

YGN 3&4 utilizes the assumptions of this Regulatory Guide for evaluating the potential radiological consequences of a LOCA, with the exception of the atmospheric dispersion model. The atmospheric dispersion model is in accordance with Regulatory Guide 1.145.

For additional information, see Subsections 2.3.4 and 15.6.5, and Appendix 15E.

REGULATORY GUIDE 1.6

Revision 0, March 1971

INDEPENDENCE BETWEEN REDUNDANT STANDBY (ONSITE) POWER SOURCES AND THEIR DISTRIBUTION SYSTEMS

YGN 3&4 complies with the regulatory position of this Regulatory Guide.

The compliance with the requirements of this guide is described in Subsections 8.3.1.1, 8.3.1.2, 8.3.2.1, and 8.3.2.2.

REGULATORY GUIDE 1.7

Revision 2, November 1978

CONTROL OF COMBUSTIBLE GAS CONCENTRATIONS IN CONTAINMENT FOLLOWING A LOSS-OF-COOLANT ACCIDENT

YGN 3&4 complies with the regulatory positions stated in this Regulatory Guide. Refer to Subsection 6.2.5 for further information.

REGULATORY GUIDE 1.8

Revision 1-R, May 1977

PERSONNEL SELECTION AND TRAINING

The personnel selection and training program described in Sections 13.1 and 13.2 complies with the intent of regulatory position set forth in this Regulatory Guide.

YGN 3&4 FSAR

REGULATORY GUIDE 1.9

Revision 2, December 1979

SELECTION, DESIGN, AND QUALIFICATION OF
DIESEL GENERATOR UNITS USED AS STANDBY (ONSITE)
ELECTRIC POWER SYSTEMS AT NUCLEAR POWER PLANTS

YGN 3&4 complies with the regulatory position of this Regulatory Guide.

The compliance with the requirements of this guide is described in Subsection 8.3.1.

REGULATORY GUIDE 1.9

Revision 3, July 1993

SELECTION, DESIGN, QUALIFICATION, AND TESTING OF
EMERGENCY DIESEL GENERATOR UNITS USED AS CLASS 1E ONSITE
ELECTRIC POWER SYSTEMS AT NUCLEAR POWER PLANTS

YGN 3&4 complies with regulatory position of this Regulatory Guide.

The compliance with the requirements of this guide described in section 8.3.1 ~~and chapter 16~~. 삭제

REGULATORY GUIDE 1.11

March 1971; Supplement, February 1972

INSTRUMENT LINES PENETRATING
PRIMARY REACTOR CONTAINMENT

YGN 3&4 complies with this Regulatory Guide. Refer to Subsection 7.1.2.15 for details.

“Delete”

REGULATORY GUIDE 1.13

Revision 1, December 1975

SPENT-FUEL STORAGE FACILITY DESIGN BASIS

YGN 3&4 complies with the regulatory positions stated in this Regulatory Guide. Refer to Subsections 9.1.2 and 9.1.3 for further information.

REGULATORY GUIDE 1.14

Revision 1, August 1975

REACTOR COOLANT PUMP FLYWHEEL INTEGRITY

YGN 3&4 complies with the intent of this Regulatory Guide.

Compliance with the guidance of this regulatory position is described in Subsection 5.4.1.

YGN 3&4 FSAR

REGULATORY GUIDE 1.16

Revision 4, August 1975

REPORTING OF OPERATING INFORMATION

Reporting of operating information complies with the Atomic Energy Act and its associated regulations of the Republic of Korea. Refer to ~~Section 16.6~~.

ITS chapter 3 5.0

REGULATORY GUIDE 1.17

Revision 1, June 1973

PROTECTION OF NUCLEAR POWER PLANTS
AGAINST INDUSTRIAL SABOTAGE

The requirements of this Regulatory Guide are incorporated into the Physical Security Plan. No comparison of the plan and this Regulatory Guide are provided in this response. Compliance is discussed further in Section 13.6.

REGULATORY GUIDE 1.20

Revision 2, May 1976*

COMPREHENSIVE VIBRATION ASSESSMENT PROGRAM FOR REACTOR INTERNALS DURING PREOPERATIONAL AND INITIAL STARTUP TESTING

YGN 3&4 complies with the intent of this Regulatory Guide.

Compliance with the guidance of this regulatory position is described in Subsection 3.9.2.

* : For replacement steam generators, Revision 3 (March 2007) of the regulatory guide is applied

REGULATORY GUIDE 1.21

Revision 1, June 1974

MEASURING, EVALUATING, AND REPORTING RADIOACTIVITY IN SOLID WASTES AND RELEASES OF RADIOACTIVE MATERIALS IN LIQUID AND GASEOUS EFFLUENTS FROM LIGHT-WATER-COOLED NUCLEAR POWER PLANTS

ITS chapter 3 5.0

As discussed in Section 11.5 and ~~Subsection 16.6~~, the program for measuring, evaluating, and reporting of effluents from the plant complies with the positions in this Regulatory Guide.

REGULATORY GUIDE 1.22

Revision 0, February 1972

PERIODIC TESTING OF PROTECTION SYSTEM ACTUATION FUNCTIONS

YGN 3&4 complies with this Regulatory Guide. Refer to Subsections 7.1.2.16 and 7.3.1.2 for further information.

REGULATORY GUIDE 1.23

Revision 0, February 1972

ONSITE METEOROLOGICAL MONITORING PROGRAMS

The onsite meteorological program described in Section 2.3.3 is designed and operated in accordance with this Regulatory Guide.

REGULATORY GUIDE 1.24

Revision 0, March 1972

ASSUMPTIONS USED FOR EVALUATING THE POTENTIAL RADIOLOGICAL CONSEQUENCES OF A PRESSURIZED WATER REACTOR RADIOACTIVE GAS STORAGE TANK FAILURE

YGN 3&4 gaseous radwaste system (GRS) design does not include pressurized radioactive waste gas storage tanks. Instead of pressurized radioactive waste gas storage tanks, radioactive waste gas charcoal delay tanks are utilized. The assumptions used in the radioactive waste gas charcoal delay tank failure analysis comply with this Regulatory Guide.

For additional information, see Subsection 15.7.1.

REGULATORY GUIDE 1.25

Revision 0, March 1972

ASSUMPTIONS USED FOR EVALUATING THE POTENTIAL RADIOLOGICAL CONSEQUENCES OF A FUEL-HANDLING ACCIDENT IN THE FUEL HANDLING AND STORAGE FACILITY FOR BOILING AND PRESSURIZED WATER REACTORS

YGN 3&4 fuel handling accident analysis complies with the recommendations of this Regulatory Guide with the following exceptions and clarifications.

A. The footnote 1.C of Regulatory Position C.1 states that average burnup for the peak assembly should be 25,000 MWD/ton or less. The batch average burnup for the peak fuel assembly is in the 44,000 MWd/MTU (4-Yr burnup) range.

B. The maximum design radial peaking factor of 1.55 is employed to be consistent with the YGN 3&4 core design peaking factors rather than this Regulatory Guide recommended radial peaking factor of 1.65 for PWR's.

Refer to subsection 15.7.4 for a discussion of this analysis.

REGULATORY GUIDE 1.26

Revision 3, February 1976

QUALITY GROUP CLASSIFICATIONS AND STANDARDS
FOR WATER-, STEAM-, AND RADIOACTIVE-WASTE CONTAINING
COMPONENTS OF NUCLEAR POWER PLANTS

YGN 3&4 complies with the regulatory positions stated in this Regulatory Guide. Refer to Subsection 3.2.2 for further information.

REGULATORY GUIDE 1.27

Revision 2, January 1976

ULTIMATE HEAT SINK FOR NUCLEAR POWER PLANTS

YGN 3&4 complies with the regulatory position set forth in this Regulatory Guide with the exception identified below. Compliance with the other requirements of this guide is described further in Subsection 9.2.5.

Regulatory Position C.1.a

"Based on regional climatological* information, select the most severe observation of the critical time period(s) for each controlling parameter or parameter combination, with substantiation of the conservation of these values for site use. The individual conditions may be combined without regard to historical occurrence.

* Climatological in this context pertains to a recent period of record at least 30 years in length.

YGN 3&4 Position

Since sufficient site specific sea water temperature data are not available, the ultimate heat sink temperature is determined based on the maximum recorded temperature from 8 years of Yellow Sea temperature profiles as listed in Table 9.2-9. 1

Justification of YGN 3&4 Position

To ensure that the ultimate heat sink temperature is conservative enough, the maximum recorded sea water temperature has been compared with the Yellow Sea water temperature profile taken from "U.S. Navy Marine Climatic Atlas of the World, Volume II, North Pacific Ocean, March 1977."

The profile indicates that the maximum recorded sea water temperature is higher than the temperature of Yellow Sea water temperature at 100 cumulative percent frequency according to U.S, Naval Data during August. Therefore the ultimate heat sink temperature of YGN 3&4 has enough conservatism.

REGULATORY GUIDE 1.28

Revision 3, August 1985*

QUALITY ASSURANCE PROGRAM REQUIREMENTS
(Design and Construction)

YGN 3&4 complies with the regulatory position of this Regulatory Guide.

Compliance with the requirements of this guide is described further in Chapter 17.

* : For replacement steam generators, Revision 4 (Jume 2010) of the regulatory guide is applied

REGULATORY GUIDE 1.29

Revision 3, September 1978

SEISMIC DESIGN CLASSIFICATION

YGN 3&4 complies with the regulatory positions stated in this Regulatory Guide. Refer to Subsections 3.2.1 and 3.10 for further information.

* : For replacement steam generators, Revision 4 (March 2007) of the regulatory guide is applied

REGULATORY GUIDE 1.30

Revision 0, August 1972

QUALITY ASSURANCE REQUIREMENTS FOR THE INSTALLATION, INSPECTION, AND TESTING OF INSTRUMENTATION AND ELECTRICAL EQUIPMENT

YGN 3&4 complies with this Regulatory Guide.

REGULATORY GUIDE 1.31

Revision 3, April 1978

CONTROL OF FERRITE CONTENT IN
STAINLESS STEEL WELD MATERIAL

YGN 3&4 complies with the intent of this Regulatory Guide.

Compliance with the guidance of this regulatory position is described in Subsection 5.2.3.

※ : For replacement steam generators, Revision 4 (October 2013) of the regulatory guide is applied

REGULATORY GUIDE 1.32

Revision 2, February 1977

CRITERIA FOR SAFETY-RELATED ELECTRIC POWER SYSTEMS
FOR NUCLEAR POWER PLANTS

YGN 3&4 complies with the regulatory position of this Regulatory Guide with the following exception and clarification:

Regulatory Position C.1.d

YGN 3&4 Complies with Regulatory Guide 1.75 for the separation of redundant sources.

Regulatory Position C.1.e

Interrupting devices actuated by fault current may be used as isolation devices, provided they are properly coordinated in accordance with Subsection 7.1.2.1 of IEEE 384-1981.

The compliance with the requirements of this guide is described in Subsections 8.1.2, 8.1.3, 8.3.1.1, 8.3.1.2, and 8.3.2.2.

REGULATORY GUIDE 1.33

Revision 2, February 1978

QUALITY ASSURANCE PROGRAM REQUIREMENTS (OPERATION)

Operation of the plant complies with Regulatory Guide 1.33, American Nuclear Society (ANSI/ANS-3.2-1982), and ANSI/ASME NQA-1-1986.

The following Regulatory Guides referenced in Regulatory Guide 1.33 are not applicable because they have been replaced by ANSI/ASME NQA-1 which is endorsed by Regulatory Guide 1.28, revision 3.

ANSI Standard	Endorsing Regulatory Guide
N45.2.6	1.58
N45.2.9	1.88
N45.2.10	1.74
N45.2.11	1.64
N45.2.12	1.144
N45.2.13	1.123
N45.2.23	1.146

REGULATORY GUIDE 1.34

Revision 0, February 1972

CONTROL OF ELECTROSLAG WELD PROPERTIES

This regulatory guide recommends controls to be applied during welding using the electroslag process. The electroslag process is not used during fabrication of any reactor coolant pressure boundary components. Therefore, the recommendations of this guide are not applicable to YGN 3&4.

YGN 3&4 FSAR

REGULATORY GUIDE 1.35

Revision 3, July 1990

INSERVICE INSPECTION OF UNGROUTED TENDONS IN PRESTRESSED CONCRETE CONTAINMENTS

YGN 3&4 complies with the regulatory position of the Regulatory Guide. Compliance with the requirements of this guide is described in Subsections 3.8.1 and ~~16.4.6.1~~. ITS chapter 3.6

REGULATORY GUIDE 1.35.1

Revision 0, July 1990

DETERMINING PRESTRESSING FORCES FOR INSPECTION OF PRESTRESSED CONCRETE CONTAINMENTS

YGN 3&4 complies with the regulatory position of the Regulatory Guide. Compliance with the requirements of this guide is described in Subsection 3.8.1.

REGULATORY GUIDE 1.36

Revision 0, February 1973

NONMETALLIC THERMAL INSULATION FOR AUSTENITIC STAINLESS STEEL

YGN 3&4 complies with the intent of applicable portions of this Regulatory Guide.

Compliance with the guidance of this regulatory position is described in Subsection 5.2.3 for the reactor vessel.

The remainder of the NSSS piping and equipment, as well as BOP piping and equipment utilizes nuclear grade blanket type insulation that has been procured and installed under a quality program in order to ensure that the chemical composition at the time of production will be limited in accordance with Regulatory Guide 1.36.

REGULATORY GUIDE 1.37*

Revision 0, March 1973

QUALITY ASSURANCE REQUIREMENTS FOR CLEANING OF FLUID SYSTEMS AND ASSOCIATED COMPONENTS OF WATER-COOLED NUCLEAR POWER PLANTS

ANSI/ASME N45.2.1** as endorsed by Regulatory Guide 1.37 shall be applied to activities occurring during the operational phase that are comparable in nature and extent to related activities occurring during construction.

* : For replacement steam generators, Revision 1 (March 2007) of the regulatory guide is applied

** : For replacement steam generators, ASME NQA-1, Part II, Subpart 2.1 is endorsed by the regulatory guide 1.37 Revision 1

REGULATORY GUIDE 1.38

Revision 2, May 1977

QUALITY ASSURANCE REQUIREMENTS
FOR PACKAGING, SHIPPING, RECEIVING,
STORAGE, AND HANDLING OF
ITEMS FOR WATER-COOLED
NUCLEAR POWER PLANTS

ANSI/ASME N45.2.2 as endorsed by Regulatory Guide 1.38 shall be applied to activities occurring during the operational phase that are comparable in nature and extent to related activities occurring during construction.

REGULATORY GUIDE 1.39

Revision 2, September 1977

HOUSEKEEPING REQUIREMENTS FOR WATER-COOLED NUCLEAR POWER PLANTS

ANSI/ASME N45.2.3 as endorsed by Regulatory Guide 1.39 shall be applied to activities occurring during the operational phase that are comparable in nature and extent to related activities occurring during construction.

REGULATORY GUIDE 1.40

Revision 0, March 1973

QUALIFICATION TESTS OF CONTINUOUS-DUTY MOTORS INSTALLED INSIDE THE CONTAINMENT OF WATER-COOLED NUCLEAR POWER PLANTS

YGN 3&4 complies with the requirements of Regulatory Guide 1.40 with the clarification to the regulatory position identified below. Applicable requirements of IEEE 334-1974 instead of IEEE 334-1971 are incorporated into the qualification tests to meet objectives set forth in this Regulatory Guide.

Regulatory Position C.1

"To the extent practicable, auxiliary equipment that will be part of the installed motor assembly should also be qualified in accordance with IEEE 334-1971."

YGN 3&4 Position

YGN 3&4 complies with the regulatory position, in that, to the extent practicable, auxiliary equipment essential to the safety function of the installed motor assembly is qualified in accordance with IEEE 334-1974.

Refer to Section 3.11 for further information on equipment qualification.

REGULATORY GUIDE 1.41

Revision 0, March 1973

PREOPERATIONAL TESTING OF REDUNDANT ONSITE ELECTRIC POWER SYSTEMS TO VERIFY POWER LOAD GROUP ASSIGNMENTS

YGN 3&4 complies with the regulatory position of this Regulatory Guide.

The compliance with the requirements of this guide is described in Subsections 8.3.1.1 and 8.3.2.2.

REGULATORY GUIDE 1.43

Revision 0, May 1973*

CONTROL OF STAINLESS STEEL WELD CLADDING OF LOW-ALLOY STEEL COMPONENTS

YGN 3&4 complies with the intent of this Regulatory Guide.

Compliance with the guidance of this Regulatory Guide is described in Subsection 5.2.3.

* : For replacement steam generators, Revision 1 (March 2011) of the regulatory guide is applied

REGULATORY GUIDE 1.44

Revision 0, May 1973

CONTROL OF THE USE OF SENSITIZED STAINLESS STEEL

YGN 3&4 complies with the intent of this Regulatory Guide with the following exception as described in Subsection 5.2.3.4.1:

> "ASTM A 708 Strauss Test is used in lieu of the ASTM A 262 Practice E, Modified Strauss Test, to demonstrate freedom from sensitization in fabricated, unstabilized, stainless steel."

※ : For replacement steam generators, Revision 1 (March 2011) of the regulatory guide is applied

REGULATORY GUIDE 1.45

Revision 0, May 1973

REACTOR COOLANT PRESSURE BOUNDARY
LEAKAGE DETECTION SYSTEMS

YGN 3&4 complies with this Regulatory Guide. Refer to Subsection 5.2.5 for further discussion.

REGULATORY GUIDE 1.47

Revision 0, May 1973

BYPASSED AND INOPERABLE STATUS INDICATION FOR NUCLEAR POWER PLANT SAFETY SYSTEMS

YGN 3&4 complies with the regulatory position of this Regulatory Guide as discussed in Subsection 7.1.2.19 and Section 7.5.

REGULATORY GUIDE 1.49

Revision 1, December 1973

POWER LEVELS OF WATER-COOLED NUCLEAR POWER PLANTS

YGN 3&4 complies with the intent of this Regulatory Guide.

Compliance with this regulatory position is described in Section 6.3 and Chapter 15.0.

REGULATORY GUIDE 1.50

Revision 0, May 1973*

CONTROL OF PREHEAT TEMPERATURE FOR WELDING
OF LOW ALLOY STEEL

YGN 3&4 complies with the intent of this Regulatory Guide with the following clarification:

"Paragraph C.1.b implies that the qualification material are an infinite heat sink that would instantaneously dissipate the heat input from the welding process. The qualification procedure consists of starting the welding at the minimum preheat temperature. Welding is continued until the maximum interpass temperature is reached. At this time the test material is permitted to cool to the minimum preheat temperature and the welding is restarted. Preheat temperatures utilized for low alloy steel are in accordance with Section III of the ASME Code. The maximum interpass temperature utilized is 500°F (260°C).

The recommendations of Regulatory Guide 1.50 are met by complying with Paragraph C.4. The soundness of all welds is verified by ASME Code acceptable examination procedures."

* : For replacement steam generators, Revision 1 (March 2011) of the regulatory guide is applied

REGULATORY GUIDE 1.52

Revision 2, March 1978

DESIGN, TESTING, AND MAINTENANCE CRITERIA FOR ENGINEERED-SAFETY-FEATURE ATMOSPHERE CLEANUP SYSTEM AIR FILTRATION AND ADSORPTION UNITS OF LIGHT-WATER-COOLED NUCLEAR POWER PLANTS

YGN 3&4 complies with the regulatory position with the following comments and exceptions keyed to paragraph numbers in Section C of the regulatory position:

Where reference is made, ANSI N-509-80 and ANSI N-510-80 will be used.

2.j Filter trains are not designed to be removable from the building as an intact unit. The size of the train precludes shipment offsite and there are no facilities for onsite disposal of the intact unit. The filter elements are removable and can be disposed of through the solid radwaste system.

2.1 Filter system housing and ducts are designed in accordance with Section 4.12 of ANSI N-509.

Duct sections will be leak tested as described in ANSI N-510 except where excluded below:

A duct section will not be subjected to quantitative measurement of leakage if one of the following conditions is satisfied.

(a) All ducts serving the protected space are located within the protected space, regardless of length.

(b) All negative pressure ducts that pass through clean interspace.

(c) All positive pressure ducts that pass through contaminated interspace with a maximum permissible concentration (MPC) within the duct (C_d) less than or equal to 1.1 times the room MPC (C_r):

$$C_d \leq 1.1\ C_r$$

(d) Positive pressure ducts that pass through a "Clean Interspace," and the effective concentration within the duct is less than 5 MPC.

(e) Negative pressure ducts that pass through a contaminated interspace with an MPC (C_r) that is no greater than 1.1 times the MPC within the duct (C_d): $C_r \leq 1.1$ (C_d)

(f) All plant vent stacks or ducts that are located outside plant buildings and no high-level or mixed-mode release credit is required to meet offsite dose limits.

Leak test acceptance criteria are based on leakage into or out of nuclear air treatment systems that may affect:

(a) Control room habitability.

(b) Plant personnel exposure during normal plant operation due to contaminated outleakage in clean spaces or clean interspaces.

(c) Plant personnel exposure due to excessive system inleakage that prevents the nuclear air treatment system from performing its design function in contaminated spaces or contaminated interspaces during plant normal, upset, or accident conditions.

(d) Offsite exposure during plant normal, upset, or accident conditions.

4.b The space provided between components is 3 feet from the front (or rear) of the components to the nearest obstacle (filter frame or other filter component). This allows 3 feet of access between components.

5.b Airflow distribution tests will be performed to ensure that the airflow through any individual filter element does not exceed 120% of the element's rated capacity.

5.c Silicone sealant or other temporary patching material will not be used in the ESF filter housings. Silicone sealant will be used, however, as a permanent sealant for HVAC ductwork.

6.a All carbon will be tested to the requirements of ASTM D3803 (1989)

6.a Laboratory tests will be performed per approved plant Technical Specifications.

Further discussions on this subject can be found in Sections 6.5, 9.4, and 12.3.

REGULATORY GUIDE 1.53

Revision 0, June 1973

APPLICATION OF THE SINGLE FAILURE CRITERIA TO NUCLEAR POWER PLANT PROTECTION SYSTEMS

YGN 3&4 complies with the guidelines for application of the single failure criteria to nuclear power plant protection systems as discussed in Subsection 7.1.2.9.

REGULATORY GUIDE 1.54

Revision 0, June 1973

QUALITY ASSURANCE REQUIREMENTS FOR PROTECTIVE COATINGS APPLIED TO WATER-COOLED NUCLEAR POWER PLANTS

For items located within the containment building, YGN 3&4 complies with the regulatory position with the following clarifications:

A. Regulatory Guide 1.54 is imposed for items located within the containment building as follows:

1. For shop priming of liner plate, structural steel and fabricated shapes.

2. For shop priming of fabricated pipes, tanks, HVAC ducts and equipment.

3. Field touchup of coated items, when the touchup area is in excess of 30 in^2.

4. For field finish painting of structural steel and equipment where called for in drawings and specifications.

5. For surfacing of concrete where indicated in drawings and specifications.

B. Regulatory Guide 1.54 is implemented as follows:

1. Only those specific coatings which are prequalified to ANSI N101.2 will be specified.

2. Surface preparation standards that have proven satisfactory in ANSI N101.2 testing will be used to govern the manufacturer's surface preparation procedures.

3. Surface profile requirements is met.

4. Application of the coating systems will be made in accordance with the coating manufacturer's detailed instructions.

5. Inspections and nondestructive testing will be made.

6. All nonconformances will be identified.

7. Certifications of compliance and/or documentation procedures will be furnished to satisfy project requirements.

C. NRC Regulatory Guide 1.54 is not imposed when:

1. The item is insulated.

2. The surface is "contained" within a cabinet or enclosure, e.g., the interior of the cab of a polar crane or the interior surfaces of ducts.

3. The field repair is less than 30 in^2 of surface area such as:

   a. Cut ends or otherwise damaged galvanized surfaces

   b. Bolt heads, nuts, and miscellaneous fasteners

   c. Damage resulting from spot, tack, or stud welding.

4. The item has a surface area less than 20 ft^2 or where special painting requirements would be impracticable, all of which will be identified in Subsection 6.1.2.

5. The surface is uncoated stainless steel or uncoated galvanized steel.

6. The coating is used for the banding of piping.

D. For the items located within the containment, where Regulatory Guide 1.54 is not imposed (see Section 6.1.2), the coating requirements includes the following:

1. The use of specific coating systems that are suitable to withstand the normal operating temperature and environment of the containment interior.

2. Specified and implemented Surface Preparation Standard SSPC-SP10.

3. Obtaining the surface profile requirements required by each coating specified.

4. Application of the coating systems in accordance with the coating manufacturer's instructions.

E. Regarding Regulatory Position C.1, ANSI N101.4-1972 will be used in conjunction with the applicable requirements of both ANSI N45.2 and ANSI/ASME NQA-1.

The compliance with the requirements of this guide is described in Subsection 6.1.2.

REGULATORY GUIDE 1.57

Revision 0, June 1973

DESIGN LIMITS AND LOADING COMBINATIONS FOR METAL PRIMARY REACTOR CONTAINMENT SYSTEM COMPONENTS

YGN 3&4 complies with the regulatory position with the following clarifications:

Piping penetration assemblies are designed by the following guidelines:

A. The portion of the primary containment penetration assembly that is part of the containment boundary, i.e., the penetration sleeve in its entire length (including the sleeve projection that forms an extension to the wall), is designed in accordance with Subsection NE, Section III of the ASME Code, augmented by the applicable provisions of Regulatory Guide 1.57.

B. The portion of the primary containment penetration assembly which consists of the head fitting (flued head) and part of the process pipeline, is designed in accordance with Subsection NB of the Code so as to satisfy stress requirements for design loadings (NB-3112), and Service Conditions (NB-3113).

Part B of the YGN 3&4 position stated above, which refers to the NB classification of the flued head and process pipe, is supported by NCA-2134 of the Code and note 3 of Regulatory Guide 1.57.

Refer to Subsection 3.8.2 for further information.

REGULATORY GUIDE 1.58

Revision 1, September 1980

QUALIFICATION OF NUCLEAR POWER PLANT INSPECTION, EXAMINATION, AND TESTING PERSONNEL

This Regulatory Guide is not applicable since it endorses ANSI/ASME N45.2.6, which was replaced for YGN 3&4 by ANSI/ASME NQA-1 (1986). Refer to the position regarding Regulatory Guide 1.33.

REGULATORY GUIDE 1.59

Revision 2, August 1977; Errata 7/30/80

DESIGN-BASIS FLOODS FOR NUCLEAR POWER PLANTS

YGN 3&4 complies with the intent of the regulatory position of this Regulatory Guide. Refer to Subsections 2.4.2, 2.4.3, 2.4.4, 2.4.5, and 2.4.6 for further information.

REGULATORY GUIDE 1.60

Revision 1, December 1973

DESIGN RESPONSE SPECTRA FOR SEISMIC DESIGN OF NUCLEAR POWER PLANTS

YGN 3&4 complies with the intent of the regulatory position of this Regulatory Guide. Refer to Subsections 2.5.2.6 and 2.5.2.7 for further information.

REGULATORY GUIDE 1.61

Revision 0, October 1973*

DAMPING VALUES FOR SEISMIC DESIGN
OF NUCLEAR POWER PLANTS

YGN 3&4 complies with the regulatory position set forth in this Regulatory Guide with the clarification and exceptions identified below. Compliance with the other requirements of this guide is described in Subsection 3.7.1 and 3.7.2.

EXCEPTION

For response spectra that are generated for piping after being filtered through the building structure, ASME Code Case N-411 damping values are used instead of the values shown in Table 1 of Regulatory Guide 1.61.

CLARIFICATION

Use of ASME Code Case N-411 was determined to be acceptable by the U.S.NRC, as documented in Regulatory Guide 1.84, Revision 24, 1986.

EXCEPTION

For cable tray systems, damping values of 10% OBE and 15% SSE are used. These values were justified by testing ("Static and Dynamic Load Test on the Cable tray for Nuclear Power Plants" by Korea Institute of Mechanics and Machinery dated April 11, 1988) and are consistent with damping values used for previously licensed Korea nuclear units.

* : For replacement steam generators, Revision 1 (March 2007) of the regulatory guide is applied

REGULATORY GUIDE 1.62

Revision 0, October 1973

MANUAL INITIATION OF PROTECTIVE ACTIONS

YGN 3&4 complies with this Regulatory Guide. Refer to Subsection 7.1.2.20 for further information.

REGULATORY GUIDE 1.63

Revision 3, February 1987

ELECTRIC PENETRATION ASSEMBLIES IN
CONTAINMENT STRUCTURES FOR LIGHT-WATER-COOLED NUCLEAR POWER PLANTS

YGN 3&4 compliance with Regulatory Guide 1.63 and IEEE 317-1983 is described below :

The electrical penetration assemblies are designed to withstand, without loss of mechanical integrity, the maximum fault current vs. time conditions which could occur as a result of single random failures of circuit overload devices. The following system features are provided to ensure compliance with this requirement of the Regulatory Guide:

Medium Voltage System

The only medium voltage loads in the containment are the reactor coolant pumps. The circuit breaker associated with each load is backed up by a second circuit breaker in series with it. This second seismically qualified circuit breaker is the bus mainfeed breaker to the reactor coolant pumps supply bus.

480-V Load Center Systems

For 480-V load center power circuits feeding loads in the containment, the circuit breaker associated with the load is backed up by a second circuit breaker in series with it. This second circuit breaker is the incoming circuit breaker to the relevant load bus. The breakers are located in the auxiliary building. The penetration withstands the available fault current for the time required for the second circuit breaker to trip in the event that the circuit breaker associated with the load fails to open.

480-V Motor Control Center Systems

For motor control center circuits feeding loads in the containment, a separate thermal-magnetic circuit breaker is provided in series with the thermal magnetic circuit breaker associated with the load. Molded case circuit breakers used in motor control centers have direct acting trips.

Low Voltage Control Systems

The majority of low voltage control circuits are of low-energy levels or are self-limiting in that the circuit resistance limits the fault current to a level which does not damage the penetration. Adequate backup protection is provided for those circuits in which the fault level at the penetration exceeds the damage limits.

Instrument Systems

The energy levels in the instrument systems are sufficiently low such that no damage can occur to the containment penetration.

The circuit overload protection system for electrical penetration assemblies meets the single failure criterion set forth in IEEE Standard 279-1971.

The overload protection systems do not conform to the on-line testability, bypassing, or manual initiation criteria of IEEE Standard 279-1971, since these criteria do not apply to these systems.

REGULATORY GUIDE 1.64

Revision 2, June 1976

QUALITY ASSURANCE REQUIREMENTS FOR THE DESIGN OF NUCLEAR POWER PLANTS

This Regulatory Guide is not applicable since it endorses ANSI/ASME N45.2.11 which has been replaced for YGN 3&4 by ANSI/ASME NQA-1 (1986). Refer to the position regarding Regulatory Guide 1.33.

REGULATORY GUIDE 1.65

Revision 0, October 1973

MATERIAL AND INSPECTIONS FOR REACTOR VESSEL CLOSURE STUDS

YGN 3&4 complies with the intent of this Regulatory Guide.

Compliance with the guidance of this regulatory position is described in Subsection 5.3.1.

REGULATORY GUIDE 1.68

Revision 2, August 1978

INITIAL TEST PROGRAMS FOR WATER-COOLED
REACTOR POWER PLANTS

The positions and guidelines of Regulatory Guide 1.68 are accepted with the exceptions and clarifications discussed in Subsection 14.2.7.

REGULATORY GUIDE 1.68.2

Revision 1, July 1978

INITIAL STARTUP TEST PROGRAM TO DEMONSTRATE REMOTE SHUTDOWN CAPABILITY FOR WATER-COOLED NUCLEAR POWER PLANTS

The position of Regulatory Guide 1.68.2 is accepted, except as follows:

Paragraph C indicates that the licensee should develop and conduct a test program for each unit. Since the YGN 3&4 units will be identical, testing on all units is unrealistic with the objectives of the test, which are:

A. Verification that the plant can be shut down from outside the control room.

B. Verification that the plant can be maintained in hot shutdown.

C. Verification of cooldown capability.

Remote shutdown testing on the unit 3 will demonstrate the above objectives. Component and preoperational testing of unit 4 and plant systems to be used in the remote shutdown panel will verify that they will function in the same manner as would be experienced on the first unit tested.

REGULATORY GUIDE 1.68.3

Revision 0, April 1982

PREOPERATIONAL TESTING OF INSTRUMENT AND CONTROL AIR SYSTEMS

YGN 3&4 complies with the intent of this Regulatory Guide. Compliance is described further in Section 14.2.

REGULATORY GUIDE 1.69

Revision 0, December 1973

CONCRETE RADIATION SHIELDS FOR NUCLEAR POWER PLANTS

ANSI N101.6-1972 was withdrawn by ANSI, and the American Concrete Institute (ACI) issued its standard ACI 349-80, "Code Requirements for Nuclear Safety-Related Concrete Structures" as well as the Commentary ACI 349R-80, which provide updated requirements with regard to the construction aspects of concrete shielding structures. ANSI/ANS-6.4-1985 was first issued as ANSI/ANS-6.4-1977 (N403) and was revised to take into account the withdrawal of ANSI N101.6-1972, the guidance provided by ACI 349-80, and recent advances in the evolution of shielding methods, data, and applications.

For these reasons, the YGN 3&4 design complies with ANSI/ANS-6.4-1985, ACI 349-80, and ACI 349R-80.

Refer to Section 12.3 for further information.

REGULATORY GUIDE 1.70

Revision 3, November 1978

STANDARD FORMAT AND CONTENT OF SAFETY ANALYSIS
REPORTS FOR NUCLEAR POWER PLANTS -
LWR EDITION

The YGN 3&4 FSAR meets the intent of the format and content recommendations of Regulatory Guide 1.70, Revision 3.

REGULATORY GUIDE 1.71

Revision 0, December 1973*

WELDER QUALIFICATION FOR AREAS OF LIMITED ACCESSIBILITY

YGN 3&4 complies with the intent of this Regulatory Guide except for the differences indicated below.

"Performance qualifications for personnel welding under conditions of limited accessibility are conducted and maintained in accordance with the requirements of ASME B&PV Code Sections III and IX. A requalification is required when (1) any of the essential variables of Section IX is changed, or (2) when authorized personnel have reason to question the ability of the welder to satisfactorily comply with the applicable requirements. Production welding is monitored for compliance with the procedure parameters, and welding qualifications are certified in accordance with Sections III and IX. Further assurance of acceptable welds of limited accessibility is afforded by the welding supervisor assigning only the most highly skilled personnel to these tasks. Finally, weld quality, regardless of accessibility, is verified by the performance of the required nondestructive examinations."

Refer also Subsection 5.2.3 and 5.3.1.

* : For replacement steam generators, Revision 1 (March 2007) of the regulatory guide is applied

REGULATORY GUIDE 1.73

Revision 0, January 1974

QUALIFICATION TESTS OF ELECTRIC VALVE OPERATORS INSTALLED INSIDE THE CONTAINMENT OF NUCLEAR POWER PLANTS

YGN 3&4 complies with the regulatory position of this Regulatory Guide.

The compliance with the requirements of this guide is described in Section 3.11 and Subsection 7.1.2.23.

REGULATORY GUIDE 1.74

Revision 0, February 1974

QUALITY ASSURANCE
TERMS AND DEFINITIONS

This Regulatory Guide is not applicable since it endorses ANSI/ASME N45.2.10, which has been replaced for YGN 3&4 by ANSI/ASME NQA-1 (1986). Refer to the position regarding Regulatory Guide 1.33.

REGULATORY GUIDE 1.75

Revision 2, September 1978

PHYSICAL INDEPENDENCE OF ELECTRIC SYSTEMS

YGN 3&4 complies with the regulatory position of this Regulatory Guide with the exceptions and/or clarifications to the regulatory positions identified and justified below:

Regulatory Position C.1

Section 3, "Isolation Device" should be supplemented as follows: "(Interrupting devices actuated only by fault current are not considered to be isolation devices within the context of this document.)"

YGN 3&4 Position

Interrupting devices actuated only by fault current may be used as isolation devices provided that the coordination criteria of IEEE 384-1981, Section 7.1.2 are met.

Justification of YGN 3&4 Position

There is no technical justification for precluding the use of Class 1E circuit breakers actuated only by fault or overload current as a circuit interrupting or isolation device.

Regulatory Position C.2

Section 3, "Raceway." Interlocked armor enclosing cable should not be construed as a "raceway."

YGN 3&4 Position

Although not a "raceway" in the same sense as a conduit or cable tray, recognition of, and design credit for the additional protection provided by the metallic jacket of interlocked armored cable should be included in the Regulatory Guide. Use of armored cable, in lieu of the separation distances stated in the Regulatory Guide, should be permitted when justified by specific testing and/or analysis, as providing the required degree of protection for Class 1E circuits against specific credible hazards.

Justification of YGN 3&4 Position

There is no technical justification for precluding the use of armored cable, in lieu of separation distances, to provide adequate isolation between Class 1E and non-Class 1E circuits and between redundant Class 1E circuits when shown to be adequate by specific testing and/or analysis.

Regulatory Position C.6

Analyses performed in accordance with Sections 4.5(3), 4.6.2, and 5.1.1.2 should be submitted as part of the Safety Analysis Report and should identify those circuits installed in accordance with these sections.

YGN 3&4 Position

The referenced analysis, when performed to justify deviation from specific requirements of standard IEEE 384-1981, is prepared on a case-by-case basis, is documented and is on permanent file, available for MOST review, but are not an integral part of the Safety Analysis Report.

Justification of YGN 3&4 Position

The YGN 3&4 position is consistent with that taken for other plant design records, e.g., routine design calculations, design document revisions, etc.

Regulatory Position C.7

Non-Class 1E instrumentation and control circuits should not be exempted from the provisions of Section 4.6.2.

YGN 3&4 Position

Low energy non-Class 1E instrumentation and control circuits are not required to be physically separated or electrically isolated from "associated" circuits provided: (a) the non-Class 1E circuits are not routed with "associated" circuits of a redundant division, and (b) they are analyzed to demonstrate that Class 1E circuits are not degraded below an acceptable level.

Justification of YGN 3&4 Position

The YGN 3&4 position is consistent with the industry consensus position regarding required separation between non-Class 1E circuits and "associated" circuits, taken in the 1977 and 1981 revisions to IEEE-384, Section 5.6(4).

Regulatory Position C.8

Section 5.1.1.1 should not be construed to imply that adequate separation of redundant circuits can be achieved within a confined space such as a cable tunnel that is effectively unventilated.

YGN 3&4 Position

Adequate separation of redundant Class 1E circuits can be achieved in areas of the plant that are effectively unventilated.

Justification of YGN 3&4 Position

There is no technical justification for precluding the routing of redundant Class 1E circuits through areas of the plant that may be "effectively unventilated" provided that adequate physical separation is provided between redundant circuits and appropriate thermal derating factors for such circuits have been incorporated into the plant design.

Regulatory Position C.9

Section 5.1.1.3 should be supplemented as follows: "(4) Cable splices in raceways should be prohibited."

YGN 3&4 Position

Cable splices, either within raceways or at the interface of raceways and equipment, etc., are permitted provided they are absolutely necessary by the plant design as indicated on the design installation drawings.

Justification of YGN 3&4 Position

There is no technical justification for precluding the use of cable splices within raceways or at their interfaces with equipment, etc., provided that they are an integral part of the plant design as indicated on the design documents. Fire propagation due to cable splicing are not foressen because cable splices are qualified in accordance with IEEE 383 and IEEE 323, and meet the flame test requirments.

Regulatory Position C.12

Pending issuance of other acceptable criteria, those portions of Section 5.1.3 (exclusive of the note following the second paragraph) that permit the routing of cables through the cable spreading area(s) and, by implication, the control room, should not be construed as acceptable. Also, Section 5.1.3 should be supplemented as follows: "Where feasible, redundant cable spreading areas should be utilized."

YGN 3&4 Position

Power cables installed in dedicated solidly enclosed metallic raceways in air (e.g., rigid steel conduit or solid cable trays with solid flush covers), may be routed through those areas designated as "cable spreading areas," where justified by analysis or other suitable means.

Justification of YGN 3&4 Position

There is no technical justification to preclude the routing of power cables through cable spreading areas when they are installed in such a manner to present no hazard to other cabling, generally of a lower energy level, within the area.

Regulatory Position C.17

Regulatory Guide Position on Section 4.6.1 "Separation from Class 1E Circuits," of IEEE Std 384 (1974)

By not modifying Section 4.6.1 of IEEE Std 384 (1974) in a Regulatory Position, the Regulatory Guide has endorsed it as stated in the IEEE standard.

YGN 3&4 Position

There is no justification for precluding the use of technically acceptable analysis to justify, on a case-by-case basis, exceptions to the generally stated criteria for separation of non-Class 1E circuits, from Class 1E circuits. When such analysis demonstrates that the following requirements are met, the non-Class 1E circuits involved need not be classified as "associated" circuits.

For specific cases, where cable termination or routing arrangements (e.g., cables leaving cable trays in free air entering equipment or passing through conduit sleeves in walls) limit the available separation distances between non-Class 1E and Class 1E cables, to less than the minimum separation applicable to redundant cables in raceways, such lesser separations are permitted provided that a documented analysis is performed to demonstrate that:

a. the non-Class 1E circuits are not routed with Class 1E circuits of a redundant division or circuits "associated" with a redundant division, and

b. the Class 1E circuits involved are not degraded below an acceptable level.

The analysis will include consideration of the potential energies of the circuits involved; the physical and electrical isolation (i.e., barriers) provided for the circuits by the cable insulation, the cable shielding, and the cable jacketing systems; the degree of environmental qualification and fire retardant characteristics of the cables; and the potential for hazards in the specific area involved.

Justification of YGN 3&4 Position

The YGN 3&4 position is consistent with the industry consensus technical position stated in the 1981 revision to IEEE-384, Section 5.6(3) with the following exception;

"Six inch separation for logic matrices, initiation and actuation circuits is not maintained nor can barriers or circuit be utilized in the PPS design. An analysis has been performed to show that the separation achieved is acceptable."

REGULATORY GUIDE 1.76

Revision 0, April 1974

DESIGN BASIS TORNADO FOR NUCLEAR POWER PLANTS

Regulatory Guide 1.76 describes the acceptable design-basis tornado (DBT) within the contiguous United States only. Therefore, the USNRC position regarding a design basis tornado does not apply to YGN 3&4. Detailed discussion regarding wind and tornado is covered in Sections 2.3 and 3.3.

REGULATORY GUIDE 1.77

Revision 0, May 1974

ASSUMPTIONS USED FOR EVALUATING A CONTROL ROD EJECTION ACCIDENT FOR PRESSURIZED WATER REACTORS

YGN 3&4 complies with the intent of the Regulatory Guide.

Compliance with the guidance of this Regulatory Guide is described in Subsection 15.4.8.

REGULATORY GUIDE 1.78

Revision 0, June 1974

ASSUMPTIONS FOR EVALUATING THE HABITABILITY OF A NUCLEAR POWER PLANT CONTROL ROOM DURING A POSTULATED HAZARDOUS CHEMICAL RELEASE

As indicated in Section 2.2, there is no prominent hazardous chemicals storage and transportation facility within 16 km of the site. Also, it is anticipated that no significant quantities of hazardous chemicals except ammonium hydroxide will be stored onsite. YGN 3&4 control room design meets the intent of Regulatory Guide 1.78. The intent is met by providing for manual isolation of the control room by the operator within 2 minutes, after the detection of ammonia by the operator in the control room. Refer to Section 6.4 for further discussion.

REGULATORY GUIDE 1.79

Revision 1, September 1975

PREOPERATIONAL TESTING OF EMERGENCY CORE COOLING SYSTEMS FOR PRESSURIZED WATER SYSTEMS

YGN 3&4 complies with the intent of this Regulatory Guide with the following clarification:

> The intent of Section C.1.c(2), Isolation Valve Test, is satisfied by opening the valves under maximum differential pressure (RCS at ambient pressure) using normal electrical power only. Conditions at the valve motor are independent of the power source for this test. The breaker response and the response of the valves to the "confirmatory open" signal is verified during the Integrated Safety Injection Actuation System Test.

Refer to Subsection 3.1.2.33 and Section 14.2 for further information.

REGULATORY GUIDE 1.81

Revision 1, January 1975

SHARED EMERGENCY AND SHUTDOWN ELECTRIC SYSTEMS FOR MULTIUNIT NUCLEAR POWER PLANTS

YGN 3&4 complies with the regulatory position of this Regulatory Guide.

The compliance with the requirements of this guide is described in Subsection 8.3.1.

REGULATORY GUIDE 1.82

Revision 1, November 1985

WATER SOURCES FOR LONG-TERM RECIRCULATION COOLING FOLLOWING A LOSS-OF-COOLANT ACCIDENT

YGN 3&4 complies with the regulatory position of this Regulatory Guide. Compliance with the requirements of this Regulatory Guide is described in Subsection 6.2.2.

YGN 3&4 FSAR

REGULATORY GUIDE 1.83

Revision 1, July 1975

INSERVICE INSPECTION OF PRESSURIZED WATER REACTOR STEAM GENERATOR TUBES

YGN 3&4 complies with the intent of applicable portions of this Regulatory Guide.

Compliance with the guidance of this regulatory position is described in Subsection 5.2.4 and ~~Chapter 16.0~~. ITS

REGULATORY GUIDE 1.84

Revision 23, September 1985*

DESIGN AND FABRICATION CODE CASE ACCEPTABILITY
ASME SECTION III DIVISION I

RESPONSE

The position of Regulatory Guide 1.84 is accepted. However, YGN 3&4 intends to use the 1986 Edition of the ASME Code, with no Addendum. Use of this edition has been found to be acceptable by the U.S. NRC. Also, the YGN 3&4 intends to use the code case N 411-1, "Alternate Damping Values for Seismic Analysis of Class 1,2 and 3 Piping Section III, Division 1." Use of this code case is also acceptable, as discussed in Revision 24 of Regulatory Guide 1.84.

Refer to Section 5.2 for further information on application of this Regulatory Guide.

* : For replacement steam generators, Revision 35 (October 2010) of the regulatory guide is applied.
In case of inconel 690(I-52 or I-52M) welding, ASME Code Case 2142-2 is used.

REGULATORY GUIDE 1.85

Revision 23, September 1985

MATERIALS CODE ACCEPTABILITY ASME SECTION III DIVISION I

RESPONSE

The position of Regulatory Guide 1.85 is accepted. However, YGN 3&4 uses the 1986 Edition of the ASME Code, with no addenda. Use of this edition has been found to be acceptable by the U.S. NRC. Also, Code Case N-474-1, "Design Stress Intensities and Yield Strength Values for UNS-N06690 With a Minimum Specified Yield Strength of 35 Ksi, Class 1 Components, Section III, Division 1," was used. This code case was approved for use by the NRC in Revision 28 of this regulatory guide.

Refer to Section 5.2 for further information on application of this Regulatory Guide.

Also, Code Case N-859 was approved to use for fittings made from ASME SB-366 UNS N04400 material. 791

※ : For replacement steam generators, this regulatory guide is incorporated into the regulatory guide 1.84

REGULATORY GUIDE 1.86

Revision 0, June 1974

TERMINATION OF OPERATING LICENSES FOR NUCLEAR REACTORS

YGN 3&4 intends to meet this Regulatory Guide.

REGULATORY GUIDE 1.88

Revision 2, October 1976

COLLECTION, STORAGE, AND MAINTENANCE OF NUCLEAR POWER PLANT QUALITY ASSURANCE RECORDS

This Regulatory Guide is not applicable since it endorses ANSI/ASME N45.2.9 which has been replaced for YGN 3&4 by ANSI/ASME NQA-1 (1986). Refer to the position regarding Regulatory Guide 1.33.

REGULATORY GUIDE 1.89

Revision 1, June 1984

ENVIRONMENTAL QUALIFICATION OF CERTAIN ELECTRICAL EQUIPMENT IMPORTANT TO SAFETY FOR NUCLEAR POWER PLANTS

Regulatory Guide 1.89, 1984, endorses IEEE 323-1974. The YGN 3&4 equipment qualification program is in compliance with IEEE 323-1974, as discussed in Section 3.11.

REGULATORY GUIDE 1.91

Revision 1, February 1978

EVALUATIONS OF EXPLOSIONS POSTULATED TO OCCUR ON TRANSPORTATION ROUTES NEAR NUCLEAR POWER PLANTS

As stated in Section 2.2, explosives are not located or transported near the YGN 3&4. Therefore, no explosion is postulated that could create a hazard at the site.

REGULATORY GUIDE 1.92

Revision 1, February 1976*

COMBINING MODAL RESPONSES AND SPATIAL
COMPONENTS IN SEISMIC RESPONSE ANALYSIS

YGN 3&4 complies with the regulatory position of this Regulatory Guide. Compliance with the requirements of this guide is described in Subsections 3.7.2 and 3.7.3.

* : For replacement steam generators, Revision 3 (September 2012) of the regulatory guide si applied

Amendment 349
2007.04.20

YGN 3&4 FSAR

REGULATORY GUIDE 1.93

Revision 0, December 1974

AVAILABILITY OF ELECTRIC POWER SOURCES

The YGN 3&4 complies with the regulatory position of this Regulatory Guide. The compliance with the requirements of this guide is described in ~~Subsection 16.3/4.8~~.
ITS chapter 1 3.8

REGULATORY GUIDE 1.94

Revision 1, April 1976

QUALITY ASSURANCE REQUIREMENTS FOR INSTALLATION, INSPECTION, AND TESTING OF STRUCTURAL CONCRETE AND STRUCTURAL STEEL DURING THE CONSTRUCTION PHASE OF NUCLEAR POWER PLANTS

KEPCO's Quality Assurance Program is in compliance with the intent of Regulatory Guide 1.94. Refer to the Quality Assurance Program.

REGULATORY GUIDE 1.95

Revision 1, January 1977

PROTECTION OF NUCLEAR POWER PLANT CONTROL ROOM OPERATORS AGAINST AN ACCIDENTAL CHLORINE RELEASE

As indicated in Section 2.2, there is no prominent chlorine storage and transportation facility within 16 km of the site. Further discussion is provided in Section 6.4.

REGULATORY GUIDE 1.97

Revision 3, May 1983

INSTRUMENTATION FOR LIGHT-WATER-COOLED NUCLEAR POWER PLANTS TO ASSESS PLANT CONDITIONS AND ENVIRONS CONDITIONS DURING AND FOLLOWING AN ACCIDENT

YGN 3&4 complies with this Regulatory Guide with the following exceptions.

1. The radioactivity concentration or radiation level in the reactor coolant is measured using grab samples and is part of the post-accident sampling system.

   The post-accident sampling system is fed from two independent Class-1E power sources. No on-line capability for monitoring radioactivity levels in the reactor coolant is provided to meet Category 1 requirements of Reg. Guide 1.97.

2. The range of the wide range boronometer is only 0-5000 ppm; CE plants cannot exceed a boron conceutration of 4,400 ppm.

Refer to Sections 3.1 and 7.5 for further information.

REGULATORY GUIDE 1.99

Revision 2, May 1988

RADIATION EMBRITTLEMENT OF REACTOR VESSEL MATERIALS

YGN 3&4 complies with the intent of this Regulatory Guide.

Compliance with this Regulatory Position is described in Section 5.3.

REGULATORY GUIDE 1.100

Revision 1, August 1977*

SEISMIC QUALIFICATION OF ELECTRIC EQUIPMENT FOR NUCLEAR POWER PLANTS

The YGN 3&4 complies with the regulatory position of this Regulatory Guide.

Compliance with the requirements of this guide is described in Section 3.10 and Subsection 7.1.2.32.

* : For replacement steam generators, Revision 3 (September 2009) of the regulatory guide is applied

REGULATORY GUIDE 1.101

Revision 2, October 1981

EMERGENCY PLANNING AND PREPAREDNESS FOR NUCLEAR POWER REACTORS

RESPONSE

Emergency planning for YGN 3&4 is in compliance with the intent of Regulatory Guide 1.101. For additional discussion, refer to Section 13.3.

REGULATORY GUIDE 1.102

Revision 1, September 1976

FLOOD PROTECTION FOR NUCLEAR POWER PLANTS

YGN 3&4 complies with intent of the regulatory position of this Regulatory Guide.

Refer to Subsections 2.4.2.3, 2.4.5.5, and 2.4.10 for further information.

REGULATORY GUIDE 1.105

Revision 2, February 1986*

INSTRUMENT SETPOINTS

YGN 3&4 complies with the regulatory position with the following exceptions keyed to paragraph numbers in the position:

Position C.5, requires locking devices on instrument setpoint adjustment mechanisms. On YGN 3&4 an individual locking feature on each instrument setpoint adjustment mechanism is provided to the extent availavable in the standard designs of the instrumentation industry in addition to the fact that accesses to the instrument setpoint adjustment mechanisms are administratively controlled by keylocked cabinet accesses or authrized software controlled program access. However, locking devices are not generally required to maintain stable instrument setpoints and we believe that setpoint stability will not be improved by providing locking devices.

Position C.6, requires documentation of the assumptions used in selecting setpoint values and the margins between the setpoints and the limiting safety system values. The documentation is to include definition of instrument setpoint drift rate and the relationship of the drift rate to testing intervals. The YGN 3&4 design conforms to this position only to the degree that setpoints are documented on the instrument data sheets along with instrument range and the maximum range of the parameter being measured. With respect to the other requirements of position C.6, generic drift rates are not generally available for any instruments since drift rates would be affected by the particular service to which the instrument was subjected. Testing intervals are set on the basis of past experience with the specific instrument types in question.

Refer to Subsection 7.1.2.33 for further information.

* : For replacement steam generators, Revision 3 (december 1999) of the regulatory guide is applied

REGULATORY GUIDE 1.106

Revision 1, March 1977

THERMAL OVERLOAD PROTECTION FOR ELECTRIC MOTORS ON MOTOR-OPERATED VALVES

YGN 3&4 complies with the regulatory position of this Regulatory Guide by using thermal overload devices set with all uncertainties in the trip setpoint resolved in favor of completing the valve operator's Safety-Related action (Regulatory Position C.2).

The compliance with the requirements of this guide is described in ~~Chapter 16~~. Subsection 13.7

REGULATORY GUIDE 1.108

Revision 1, August 1977 (supplemented September 1977)

PERIODIC TESTING OF DIESEL GENERATOR UNITS USED AS ONSITE ELECTRIC POWER SYSTEMS AT NUCLEAR POWER PLANTS

YGN 3&4 complies with the regulatory position of this Regulatory Guide.

The compliance with the requirements of this guide is described in Subsections 8.3.1. 153

REGULATORY GUIDE 1.109

Revision 1, October 1977

CALCULATION OF ANNUAL DOSES TO MAN FROM ROUTINE RELEASES OF REACTOR EFFLUENTS FOR THE PURPOSE OF EVALUATING COMPLIANCE WITH 10 CFR 50, APPENDIX I

YGN 3&4 utilizes the methodology of this Regulatory Guide in estimating potential annual doses to maximum hypothetical receptors from routine releases of reactor effluents in conjunction with site-specific parameters.

Both liquid and gaseous pathways are modeled. For additional information, see Subsections 11.2.3 and 11.3.3 and ~~Chapter 16~~ ITS.

REGULATORY GUIDE 1.110

Revision 0, March 1976

COST-BENEFIT ANALYSIS FOR
RADWASTE SYSTEMS FOR
LIGHT-WATER-COOLED NUCLEAR POWER REACTORS

Since the liquid and gaseous radwaste systems follow, with the increased capacity and redundancy, the design concepts of YGN 1&2 which had been designed to satisfy the requirements of Appendix I to 10 CFR 50, the radwaste systems of YGN 3&4 can be considered to be designed with a favorable cost-benefit ratio. Furthermore, since the population dose within 50 miles (80 km) of the site due to the liquid and gaseous waste effluents routinely released is estimated to be only a few man-rem as described in Section 11.2 and 11.3, there is no demonstrated technology to reduce the population dose further with a reasonable cost-benefit ratio.

Therefore, a cost-benefit analysis as identified in this Regulatory Guide was considered to be unnecessary.

REGULATORY GUIDE 1.111

Revision 1, July 1977

METHODS FOR ESTIMATING ATMOSPHERIC TRANSPORT AND DISPERSION OF GASEOUS EFFLUENTS IN ROUTINE RELEASES FROM LIGHT-WATER-COOLED REACTORS

YGN 3&4 utilizes the methods of this Regulatory Guide in estimating atmospheric transport and dispersion of gaseous effluents in routine releases.

For additional information, see Subsection 2.3.5.

REGULATORY GUIDE 1.112

Revision 0-R, May 1977

CALCULATION OF RELEASES OF RADIOACTIVE MATERIALS IN GASEOUS AND LIQUID EFFLUENTS FROM LIGHT-WATER-COOLED POWER REACTORS

YGN 3&4 utilizes the methodology of this Regulatory Guide when calculating releases of radioactive materials in gaseous and liquid effluents.

For additional information, see Subsections 11.2.3 and 11.3.3.

REGULATORY GUIDE 1.113

Revision 1, April 1977

ESTIMATING AQUATIC DISPERSION OF EFFLUENTS FROM ACCIDENTAL AND ROUTINE REACTOR RELEASES FOR THE PURPOSE OF IMPLEMENTING APPENDIX I

YGN 3&4 utilizes the methodology of this Regulatory Guide in estimating aquatic dispersion of effluents from accidental and routine releases, with site-specific parameters.

In Appendix A of this Regulatory Guide, there are a number of methods presented to model the release of liquid effluents into ocean coastal waters. For YGN 3&4, a steady plane source discharge Gaussian model is used.

For further information, see Section 11.2.

YGN 3&4 FSAR

REGULATORY GUIDE 1.114

Revision 1, November 1976

GUIDANCE ON BEING OPERATOR AT THE CONTROLS OF A NUCLEAR POWER PLANT

RESPONSE

The regulatory positions of this Regulatory Guide are met. Refer to Subsection 13.1.3 and ~~Section 16.6~~.
ITS chapter 3

REGULATORY GUIDE 1.115

Revision 1, July 1977

PROTECTION AGAINST LOW-TRAJECTORY TURBINE MISSILES

YGN 3&4 complies with the regulatory position of this Regulatory Guide. Compliance with the requirements of this guide is described in Section 3.5.

REGULATORY GUIDE 1.116

Revision 0-R, May 1977

QUALITY ASSURANCE REQUIREMENTS FOR INSTALLATION, INSPECTION, AND TESTING OF MECHANICAL EQUIPMENT AND SYSTEMS

YGN 3&4 complies with the regulatory position of this Regulatory Guide with the following clarifications:

a. The requirements for installation, inspection, and testing will be those included in ANSI/ASME NQA-2, Quality Assurance Requirements for Nuclear Power Plants, 1986 Edition, Part 2.8, which incorporates the requirements of ANSI N45.2.8-1975 endorsed by Regulatory Guide 1.116. ANSI N45.2.8-1975 was withdrawn after issue of the Regulatory Guide. Regulatory positions C.1, C.2 and C.3 will apply to the corresponding sections of ANSI/ASME NQA-2.

b. Reference: Section 2.1 of ANSI/ASME NQA-2-1986, Part 2.8, Planning and Procedures. The required planning is frequently performed on a generic basis for application to many installations on one or more projects. This results in standard procedures or plans for installation, inspection, and testing that meet the requirements of the Standard.

   Individual plans for each item or system are not normally prepared unless the work operations are unique. However, standard procedures or plans are reviewed for applicability in each case. Installation plans or procedures are also limited in scope to those actions or activities that are essential to maintain or achieve required quality.

c. Reference: Section 2.2 of ANSI/ASME NQA-2-1986, Part 2.8, Prerequisites. Item e.6 is interpreted to mean that any work performed without an approved design change shall not be considered complete and acceptable for its intended use until the change is approved, and that the intent of this item will be satisfied provided that such work is performed only with approved procedures and that the activities and the results are documented. Evidence of design change approval shall be required prior to placing the affected item in service.

d. Reference: Section 3.3 of ANSI/ASME NQA-2-1986, Part 2.8, Processes and Procedures. The terms "work site," and "site" are interpreted to mean the same as "construction site;" when applied to documents, these may be at the central office or work area document control station. The term "installation area" is interpreted to mean the immediate proximity of location where work is to be performed.

e. Reference: Section 3.5 of ANSI/ASME NQA-2-1986, Part 2.8, Site Conditions. Item a is applied only if subsequent correction of adjacent nonconformances could damage the item being installed.

f. Reference: Section 4.5 of ANSI/ASME NQA-2-1986, Part 2.8, Care of Items. The constructor or construction manager is assumed to have authority and is the "responsible organization" for temporary usage of equipment or facilities unless specifically prohibited by contract or in writing from the owner. All other conditions and considerations for temporary use in this section are applied.

g. Reference: Section 5 of ANSI/ASME NQA-2-1986, Part 2.8, Installed Systems Inspection and Tests. For the purposes of functional tests addressed by this standard, YGN 3&4 defines completed systems as any system, or portion or component thereof, on which construction is sufficiently complete to allow the required testing, and on which further or adjacent construction will not render the results of such testing invalid or indeterminate.

h. Reference: Sections 5.2 and 5.4 of ANSI/ASME NQA-2-1986, Part 2.8, Preoperational Testing and Hot Functional Tests. For application of the provisions of these sections to preoperational and startup testing, YGN 3&4 position on Regulatory Guide 1.68, "Preoperational and Initial Startup Test Programs for Water-Cooled Power Reactors," Revision 2, shall take precedence where there is a conflict or difference.

REGULATORY GUIDE 1.117

Revision 1, April 1978

TORNADO DESIGN CLASSIFICATION

YGN 3&4 complies with the intent of the Regulatory Position of this Regulatory Guide. The YGN 3&4 are designed to withstand the effects of typhoon winds in lieu of a tornado. Detailed discussions regarding compliance with the requirements of this guide but with respect to typhoon wind loads are presented in Section 3.3.

REGULATORY GUIDE 1.118

Revision 2, June 1978

PERIODIC TESTING OF ELECTRIC POWER AND PROTECTION SYSTEMS

YGN 3&4 complies with the regulatory position of this Regulatory Guide.

The compliance with the requirements of this guide is described in Subsection 7.1.2.7.

REGULATORY GUIDE 1.120

Draft (Revision 1), November 1977

FIRE PROTECTION GUIDELINES FOR NUCLEAR POWER PLANTS

Not applicable. This Regulatory Guide was issued as a draft and has not been issued for use. YGN 3&4 complies with the intent of USNRC Branch Technical Position CMEB 9.5-1.

YGN 3&4 FSAR

REGULATORY GUIDE 1.121

August, 1976

BASES FOR PLUGGING DEGRADED PWR
STEAM GENERATOR TUBES

YGN 3&4 complies with the intent of the applicable portions of this Regulatory Guide.

Compliance with the guidance of this regulatory position is described in ~~Chapter 16~~. ITS

REGULATORY GUIDE 1.122

Revision 1, February 1978

DEVELOPMENT OF FLOOR DESIGN RESPONSE SPECTRA FOR SEISMIC DESIGN OF FLOOR SUPPORTED EQUIPMENT OR COMPONENTS

YGN 3&4 complies with the regulatory position of this Regulatory Guide.

Compliance with the requirements of this guide is described in Subsection 3.7.2.

REGULATORY GUIDE 1.123

Revision 1, July 1977

QUALITY ASSURANCE REQUIREMENTS FOR CONTROL OF PROCUREMENT OF ITEMS AND SERVICES FOR NUCLEAR POWER PLANTS

This Regulatory Guide is not applicable since it endorses ANSI/ASME N45.2.13, which was replaced for YGN 3&4 by ANSI/ASME NQA-1 (1986). Refer to the position regarding Regulatory Guide 1.33.

REGULATORY GUIDE 1.124

Revision 1, January 1978

SERVICE LIMITS AND LOADING COMBINATIONS FOR CLASS-1 LINEAR-TYPE COMPONENT SUPPORTS

YGN 3&4 complies with the intent of this Regulatory Guide.

Compliance with the guidance of this Regulatory Guide is described in Subsection 3.9.3.

REGULATORY GUIDE 1.125

Revision 1, October 1978

PHYSICAL MODELS FOR DESIGN AND OPERATION OF HYDRAULIC STRUCTURES AND SYSTEMS FOR NUCLEAR POWER PLANTS

Physical models were not utilized for safety-related structures to predict the action or interaction of surface waters with the hydraulic structure.

The design for the hydraulic structures and systems of YGN 3&4 are based upon standard design practice and existing designs. Therefore this Regulatory Guide is not applicable to YGN 3&4.

REGULATORY GUIDE 1.126

Revision 1, March 1978

AN ACCEPTABLE MODEL AND RELATED STATISTICAL METHODS FOR THE ANALYSIS OF FUEL DENSIFICATION

YGN 3&4 complies with the intent of this Regulatory Guide.

Compliance with the guidance of this regulatory position is described in Subsection 4.2.1.2.4.3.

REGULATORY GUIDE 1.127

Revision 1, March 1978

INSPECTION OF WATER-CONTROL STRUCTURES ASSOCIATED WITH NUCLEAR POWER PLANTS

Water-control structures are not used in the YGN 3&4 design and, therefore, this Regulatory Guide is not applicable.

REGULATORY GUIDE 1.128

Revision 1, October 1978

INSTALLATION DESIGN AND INSTALLATION OF LARGE LEAD STORAGE BATTERIES FOR NUCLEAR POWER PLANTS

YGN 3&4 complies with the regulatory position of this Regulatory Guide with the exceptions and/or clarifications to the regulatory positions identified and justified below:

Regulatory Position C.1

In Subsection 4.1.4 "Ventilation," instead of the second sentence, the following should be used:

"The ventilation system shall limit hydrogen concentration to less than 2% by volume at any location within the battery area."

YGN 3&4 Position

The ventilation requirements set forth in IEEE Std. 484-1981 are adequate.

Justification of YGN 3&4 Position

IEEE 484-1981 requires that the ventilation system limit hydrogen accumulation to less than 2% of the total volume of the battery area. This regulatory position would require that hydrogen accumulation be limited to less than 2% at any location within the battery area. The ventilation requirements as set forth in IEEE 484-1981 are entirely adequate. The "2% at any location" requirement would be almost impossible to verify and might even require the installation of ducts, vanes, and/or auxiliary fans so as to ensure that every location is thoroughly purged.

The battery area ventilation system is designed to maintain the hydrogen concentration below 2% with a "run-away" charger (i.e., a charger delivering its full-rated output into a fully-charged battery, thereby causing gasing of all cells). Thus, any significant hydrogen buildup in the battery area would require two failures (a failure of the ventilation system, and a failure of the charger), both of which will be annunciated in the main control room. Alarms provided for these two failures are adequate to insure that there will be no significant accumulation of hydrogen in the battery area.

Regulatory Position C.2

In Subsection 4.2.1, "Location," Item 1, the general requirement that the battery be protected against fire should be supplemented with the applicable recommendations in Regulatory Guide 1.120, "Fire Protection Guidelines for Nuclear Power Plants."

YGN 3&4 Position

The reference to Regulatory Guide 1.120 is inappropriate because this Regulatory Guide was issued only for comment and not issued for use.

Justification of YGN 3&4 Position

The battery location and protection against fire will be in accordance with the intent of USNRC Chemical Engineering Branch (CMEB), Branch Technical Position 9.5-1. The location and fire protection requirements set forth in IEEE 484-1981 are adequate.

YGN 3&4 FSAR

REGULATORY GUIDE 1.129

Revision 1, February 1978

MAINTENANCE, TESTING, AND REPLACEMENT OF
LARGE LEAD STORAGE BATTERIES FOR NUCLEAR POWER PLANTS

The YGN 3&4 complies with the regulatory position of this Regulatory Guide. The compliance with the requirements of this guide is described in ~~Subsection 16.3/4.8.2~~.

ITS chapter 1 3.8

REGULATORY GUIDE 1.130

Revision 1, October 1978

SERVICE LIMITS AND LOADING COMBINATIONS FOR
CLASS-1 PLATE-AND-SHELL-TYPE
COMPONENT SUPPORTS

YGN 3&4 complies with the intent of this Regulatory Guide.

Compliance with the guidance of this regulatory position is described in Subsection 3.9.3.

REGULATORY GUIDE 1.131

Revision 0, August 1977

QUALIFICATION TESTS OF ELECTRIC CABLES
FIELD SPLICES AND CONNECTIONS FOR
LIGHT WATER-COOLED NUCLEAR POWER PLANTS

YGN 3&4 complies with the regulatory position with the following comments and exceptions keyed to paragraph numbers in the position:

1. Regulatory Position C.1

   The position states that in lieu of Section 1.3.4.2.3 of IEEE-383, "Other Design Basis Events," the following should be used: "The remainder of the complete spectrum of design basis events (e.g., events such as a steam-line break) shall be considered in case they represent different types of more severe hazards to cable operation."

   YGN 3&4 Position

   All safety-related cables are qualified for the anticipated environments detailed in Section 3.11 including the design basis accidents addressed in Section 6.2.

2. Regulatory Position C.10

   The position states that in lieu of the first sentence of Section 2.5.4.4.1 of IEEE-383, the following should be used: The ribbon gas burner shall be mounted horizontally such that the flame impinges on the specimen midway between the tray rungs and so that the burner face is in front of and 4 in. from the cable and approximately two feet above the bottom of the tray.

   YGN 3&4 Position

   YGN 3&4 complies with Section 2.5.4.4.1 of IEEE-383.

3. Regulatory Position C.11

   The position states that in lieu of Section 2.5.4.4.3 of IEEE-383, the following should be used: "Flame size will normally be achieved when the propane flow is 27.8 standard ft^3 per hour and the air flow is 139 standard ft^3 per hour."

YGN 3&4 Position

Flame size is achieved using the schematic arrangement and pressure as set forth in IEEE-383, Subsection 2.5.4.4.3.

REGULATORY GUIDE 1.132

Revision 1, March 1979

SITE INVESTIGATION FOR FOUNDATION OF NUCLEAR POWER PLANTS

YGN 3&4 complies with the regulatory position of this Regulatory Guide, as discussed in Section 2.5.

REGULATORY GUIDE 1.133

Revision 1, May 1981

LOOSE PART DETECTION PROGRAM FOR THE PRIMARY SYSTEM OF LIGHT WATER COOLED REACTORS

YGN 3&4 complies with the intent of this Regulatory Guide.

Compliance with the guidance of this regulatory position is described in Subsection 7.1.2.25.

REGULATORY GUIDE 1.134

Revision 1, March 1979

MEDICAL EVALUATION OF NUCLEAR POWER PLANT PERSONNEL REQUIRING OPERATOR LICENSES

The intent of Regulatory Guide 1.134 is met. The applicants for Reactor Operator or Senior Reactor Operator are required to submit a certificate of medical examination by a general hospital to the regulatory authority in accordance with the applicable Republic of Korea (R.O.K.) regulation.

REGULATORY GUIDE 1.135

Revision 0, September 1977

NORMAL WATER LEVEL AND DISCHARGE AT NUCLEAR POWER PLANTS

YGN 3&4 complies with the regulatory position of this Regulatory Guide, as discussed in Section 2.4.

REGULATORY GUIDE 1.136

Revision 2, June 1981

MATERIALS, CONSTRUCTION, AND TESTING OF CONCRETE CONTAINMENTS

YGN 3&4 complies with the regulatory position of this Regulatory Guide. Compliance with the requirements of this guide is described in Subsection 3.8.1. Regulatory position C.2 is not applicable since grouted tendon systems are not used on YGN 3&4.

REGULATORY GUIDE 1.137

Revision 1, October 1979

FUEL-OIL SYSTEMS FOR STANDBY DIESEL GENERATORS

YGN 3&4 complies with the requirements of this Regulatory Guide with the exception that certain components or parts that are not available as ASME Section III items are classified as safety-related, non-ASME. This exception is taken because some fuel oil components are no longer manufactured by ASME Section III suppliers. All Safety-Related components maintain seismic qualification and conform to Regulatory Guide 1.26 and 10 CFR 50 Appendix B. All safety-related piping remains ASME Section III piping.

Refer to Sections 3.2 and 9.5 for further information.

REGULATORY GUIDE 1.138

Revision 0, April 1978

LABORATORY INVESTIGATIONS OF SOILS FOR
ENGINEERING ANALYSIS AND DESIGN OF NUCLEAR POWER PLANTS

The YGN 3&4 complies with the regulatory position of this Regulatory Guide, as discussed in Section 2.5.

REGULATORY GUIDE 1.139

Draft, May 1978

GUIDANCE FOR RESIDUAL HEAT REMOVAL

Not applicable. This draft Regulatory Guide was issued for comment and has not been issued for use.

REGULATORY GUIDE 1.140

Revision 1, October 1979

DESIGN, TESTING, AND MAINTENANCE CRITERIA FOR NORMAL VENTILATION EXHAUST SYSTEM AIR FILTRATION AND ADSORPTION UNITS OF LIGHT-WATER-COOLED NUCLEAR POWER PLANTS

The design of the non-safety-related filter systems meets the requirements of this guide, except as noted below. The exceptions are keyed to paragraph numbers in the Regulatory Position.

ANSI N-509-1980 and ANSI N-510-1980 Standards are used instead of ANSI N-509-76 and ANSI-509-1976 wherever referenced in the Regulatory Guide.

2f - Filter System Housing and Ducts are designed in accordance with Section 4.12 of ANSI N-509.

3f - Duct sections will be leak tested as described in ANSI 510 except where excluded below:

A duct section will not be subjected to quantitative measurement of leakage if one of the following conditions is satisfied.

(a) All ducts serving the protected space, are located within the protected space, regardless of length.

(b) All negative pressure ducts that pass through clean interspace.

(c) All positive pressure ducts that pass through contaminated interspace with a maximum permissible concentration (MPC) within the duct (C_d) less than or equal to 1.1 times the room MPC (C_r):

$$C_d \leq 1.1\ C_r$$

(d) Positive pressure ducts that pass through a "Clean Interspace," and the effective concentration within the duct is less than 5 MPC.

(e) Negative pressure ducts that pass through a contaminated interspace with an MPC (C_r) that is no greater than 1.1 times the MPC within the duct (C_d):

$$C_r \leq 1.1\ (C_d)$$

(f) All plant vent stacks or ducts that are located outside plant buildings and no high-level or mixed-mode release credit is required to meet offsite dose limits.

Leak test acceptance criteria are based on leakage into or out of nuclear air treatment systems that may affect:

(a) Control Room Habitability.

(b) Plant personnel exposure during normal plant operation due to contaminated outleakage in clean spaces or clean interspaces.

(c) Plant personnel exposure due to excessive system inleakage that prevents the nuclear air treatment system from performing its design function in contaminated spaces or contaminated interspaces during plant normal, upset, or accident conditions.

(d) Offsite exposure during plant normal, upset, or accident conditions.

3a- The components of the heaters were manufactured and assembled as per Section 5.5 of ANSI N509-76, similar to the requirements of heaters in Safety-Related filter systems, but the traceability of the components is not established as it would be in the case of safety-related heaters. Thus, no complete qualification program is done.

5b- Airflow distribution and air-aerosol mixing tests will not be performed on the non-entry type filter units. Airflow distribution tests will be performed on all entry-type filter trains to ensure that the airflow through any individual filter element does not exceed 120% of the element's rated capacity.

5c- Silicone sealants or other temporary patching material will not be used in the non-ESF filter housings. Silicone sealant is used, however, as a permanent sealant for HVAC ductwork.

6a- All carbon will be tested to the requirements of ASTM D3803-C1989.

HVAC Systems that are subject to these test requirements are described in Section 9.4.

REGULATORY GUIDE 1.141

Revision 0, April 1978

CONTAINMENT ISOLATION PROVISIONS FOR FLUID SYSTEMS

YGN 3&4 complies with the regulatory position with the following clarifications:

There are differences between various figures in Appendix "B" of ANSI/ANS 56.2 and containment isolation features for various systems of YGN 3&4 as shown in Subsection 6.2.4. Appendix "C" of ANSI/ANS 56.2 pertains to diagram legend and symbols, to which the YGN 3&4 conforms with minor exceptions. Appendix "D" of ANSI/ANS 56.2 pertains to a valve maintenance program which YGN 3&4 does not agree to implement. YGN 3&4 agrees with the exceptions which the guide has taken to ANSI/ANS 56.2

REGULATORY GUIDE 1.142

Revision 1, October 1981

SAFETY-RELATED CONCRETE STRUCTURES FOR NUCLEAR POWER PLANTS (OTHER THAN REACTOR VESSELS AND CONTAINMENTS)

YGN 3&4 complies with the recommendations of this Regulatory Guide.

Compliance with the recommendations of this guide is discussed in Subsection 3.8.4.

REGULATORY GUIDE 1.143

Revision 1, October 1979

DESIGN GUIDANCE FOR RADIOACTIVE WASTE MANAGEMENT SYSTEMS, STRUCTURES AND COMPONENTS INSTALLED IN LIGHT-WATER-COOLED NUCLEAR POWER PLANTS

YGN 3&4 complies with the requirements of this guide with the following exceptions and clarifications:

1. Reference: Section B, Page 2, First Paragraph, First Sentence - It is clarified that the radwaste systems for YGN 3&4 are considered to begin with and include the interface valves.

2. Reference: Section B, Page 2, First Paragraph, Second Sentence - It is clarified that the radwaste system for YGN 3&4 terminates at the end of the pipe containing the last isolation valve before the point of controlled discharge to the service water discharge line, or at the end of the pipe containing the last isolation valve before the cycle condensate storage tank, or at the point of storage of packaged solid waste prior to shipment offsite to licensed burial ground.

3. Reference: Paragraphs C.1.1.2, C.2.1.2, C.3.1.2 - Materials for pressure-retaining components conform to the requirements of ASTM Specifications.

4. Reference: Paragraph 4.3 - It is clarified that the scope of radwaste system pressure testing includes all pressure-retaining components, appurtenances, and completed systems. Bolts, studs, washers, gaskets and possible localized instances of pump and valve packing are exempted from the pressure test. This is consistent with ASME Section III NB6000 and ANSI B31.1 (1983 edition). The gaseous waste system will be pneumatically tested at a minimum of 75 psig for no less than 30 minutes.

   Portions of the radioactive waste managements system may contain polypropylene lined steel pipe and valves. These pipes and valves would be in the demineralizer subsystem because of the superior corrosion resistance of polypropylene to chemicals. This portion of the subsystem will only be inservice leak tested at normal operating pressure.

It is clarified that the lines providing water to the shaft seals of pumps in radioactive waste service shall be exempted from consideration as a process system under this regulatory guide.

Descriptions of the radioactive waste management systems are presented in Subsections 11.2.2, 11.3.2, and 11.4.2 for liquid, gaseous, and solid waste systems, respectively. Compliance with the seismic design criteria of this Regulatory Guide is discussed in Section 3.2.

REGULATORY GUIDE 1.144

Revision 1, September 1980

AUDITING OF
QUALITY ASSURANCE PROGRAMS FOR
NUCLEAR POWER PLANTS

This Regulatory Guide is not applicable since it endorses ANSI/ASME N45.2.12, which was replaced for YGN 3&4 by ANSI/ASME NQA-1 (1986). Refer to the position regarding Regulatory Guide 1.33.

REGULATORY GUIDE 1.145

Revision 1, February 1982

ATMOSPHERIC DISPERSION MODELS FOR POTENTIAL ACCIDENT CONSEQUENCE ASSESSMENTS AT NUCLEAR POWER PLANTS

YGN 3&4 use the methodology of this Regulatory Guide with site specific meteorological data to determine the appropriate atmospheric dispersion model. For additional information, see Subsection 2.3.4.

REGULATORY GUIDE 1.146

Revision 0, August 1980

QUALIFICATION OF QUALITY ASSURANCE PROGRAM
AUDIT PERSONNEL FOR
NUCLEAR POWER PLANTS

This Regulatory Guide is not applicable since it endorses ANSI/ASME N45.2.23 which was replaced for YGN 3&4 by ANSI/ASME NQA-1 (1986). Refer to the position regarding Regulatory Guide 1.33.

REGULATORY GUIDE 1.147

Revision 4, September 1985*

INSERVICE INSPECTION CODE CASE ACCEPTABILITY
ASME SECTION XI DIVISION I

YGN 3&4 complies with the regulatory position. For further discussion of ISI refer to Subsection 5.2.4 and Section 6.6.

※ : For replacement steam generators, Revision 16 (October 2010) of the regulatory guide is applied

REGULATORY GUIDE 1.148

Revision 0, March 1981

FUNCTIONAL SPECIFICATION FOR ACTIVE VALVE ASSEMBLIES IN SYSTEMS IMPORTANT TO SAFETY IN NUCLEAR POWER PLANTS

YGN 3&4 complies with the intent of the regulatory position. Compliance is discussed in Subsection 3.9.3 and Section 3.10.

REGULATORY GUIDE 1.149

Revision 1, April 1987

NUCLEAR POWER PLANT SIMULATORS FOR USE IN OPERATOR TRAINING

YGN 3&4 complies with the intent of this Regulatory Guide, as is discussed in Appendix 1B and Subsection 13.2.

REGULATORY GUIDE 1.150

Revision 1, February 1983

ULTRASONIC TESTING OF REACTOR VESSEL WELDS DURING PRESERVICE AND INSERVICE EXAMINATIONS

YGN 3&4 complies with the intent of this Regulatory Guide.

The compliance with the requirements of this guide is described in Subsection 5.3.1.

REGULATORY GUIDE 1.151

Revision 0, July 1983

INSTRUMENT SENSING LINES

YGN 3&4 complies with this Regulatory Guide. The following clarification is noted for the inspection and certification of ASME Section III instrument sensing lines;

> The inspection and certification of ASME Section III instrument sensing lines which consist of piping are covered by the code.

REGULATORY GUIDE 1.152

Revision 0, November, 1985

CRITERIA FOR PROGRAMMABLE DIGITAL COMPUTER SYSTEM SOFTWARE IN SAFETY RELATED SYSTEMS OF NUCLEAR POWER PLANTS

YGN 3&4 is in accordance with the regulatory position of this Regulatory Guide.

Core Protection Calculator (CPC) software is developed and tested in accordance with Regulatory Guide 1.152 as described by CEN-39(A)-P, "CPC Protection Algorithm Software Change Procedures" and as described in Subsection 7.1.2.27.

Similarly, Interposing Logic System (ILS) software is developed and tested in
accordance with Regulatory Guide 1.152 as discussed in Subsection 7.1.2.27 and 1
the ILS function is described in Subsection 7.3.1.1.2.1.

REGULATORY GUIDE 1.153

Revision 0, December 1985

CRITERIA FOR POWER, INSTRUMENTATION, AND CONTROL PORTIONS OF SAFETY SYSTEMS

YGN 3&4 complies with the regulatory position of this Regulatory Guide. The compliance with the requirements of this guide is described in Subsection 7.1.2.13.

REGULATORY GUIDE 1.154

January 1987

FORMAT AND CONTENT OF PLANT SPECIFIC PRESSURIZED THERMAL SHOCK SAFETY ANALYSIS REPORTS

This Regulatory Guide is not applicable to YGN 3&4.

See also Regulatory Guide 1.2.

REGULATORY GUIDE 1.155

Revision 1, August 1988

STATION BLACKOUT

YGN 3&4 complies with the regulatory position of this Regulatory Guide.

The compliance with the requirements of this guide is described in Subsections 3.1.2.5, 8.3.1, and 9.5.12.

REGULATORY GUIDE 1.156

November 1987

ENVIRONMENTAL QUALIFICATION OF CONNECTION ASSEMBLIES FOR NUCLEAR POWER PLANTS

This Regulatory Guide is not applicable to YGN 3&4.

REGULATORY GUIDE 1.157

May 1989

BEST-ESTIMATE CALCULATION OF EMERGENCY CORE COOLING SYSTEM PERFORMANCE

Regulatory Guide is not applicable to YGN 3&4. All emergency core cooling analyses performed for YGN 3&4 are based on the more stringent requirements of Appendix K. Results of these ECCS analyses are presented in Section 6.3.

REGULATORY GUIDE 8.2

Revision 0, February 1973

GUIDE FOR ADMINISTRATIVE PRACTICES IN RADIATION MONITORING

The position of Regulatory Guide is accepted. Refer to Subsections 12.1.1, 12.3.4, 12.5 and 13.2.

REGULATORY GUIDE 8.3

Revision 0, February 1973

FILM BADGE PERFORMANCE CRITERIA

The position of Regulatory Guide 8.3 is accepted. Refer to Subsection 12.5.3.6.

REGULATORY GUIDE 8.4

Revision 0, February 1973

DIRECT READING AND INDIRECT READING POCKET DOSIMETERS

The position of Regulatory Guide is accepted. Refer to Subsections 12.5.2.4 and 12.5.3.6.

REGULATORY GUIDE 8.5

Revision 1, March 1981

CRITICALITY AND OTHER INTERIOR EVACUATION SIGNALS

The position of Regulatory Guide is accepted. Refer to Subsection 12.5.3.3.3.

REGULATORY GUIDE 8.7

Revision 0, May 1973

OCCUPATIONAL RADIATION EXPOSURE RECORDS SYSTEM

The position of Regulatory Guide is accepted, and is discussed in Section 12.5.

REGULATORY GUIDE 8.8

Revision 3, June 1978

INFORMATION RELEVANT TO ENSURING THAT OCCUPATIONAL RADIATION EXPOSURES AT NUCLEAR POWER STATIONS WILL BE AS LOW AS REASONABLY ACHIEVABLE

YGN 3&4 complies with the intent of the regulatory position of this Regulatory Guide. Refer to Sections 12.1, 12.3, and 12.5 for further information.

REGULATORY GUIDE 8.9

Revision 0, September 1973

ACCEPTABLE CONCEPTS, MODELS, EQUATIONS, AND ASSUMPTIONS FOR A BIOASSAY PROGRAM

The position of Regulatory Guide is accepted. Refer to Section 12.5.

REGULATORY GUIDE 8.10

Revision 1-R, May 1977

OPERATING PHILOSOPHY FOR MAINTAINING OCCUPATIONAL RADIATION EXPOSURES AS LOW AS IS REASONABLY ACHIEVABLE

The position of Regulatory Guide is accepted. Refer to Subsection 12.3.1, and Sections 12.5 and 13.2.

REGULATORY GUIDE 8.12

Revision 1, January 1981

CRITICALITY ACCIDENT ALARM SYSTEMS

YGN 3&4 takes exception to the requirements of this Regulatory Guide as allowed by Regulatory Position C.1. The design of the spent fuel and new fuel storage areas precludes the possibility of inadvertent criticality, as discussed in Section 9.1. In addition, reactor core criticality will be prevented during refueling operations (see Section 9.1). Therefore, criticality monitors are not required.

The above discussion documents the YGN 3&4 request for exemption from 10 CFR 70.24, as allowed by Regulatory Position C.1 of this Regulatory Guide.

REGULATORY GUIDE 8.13

Revision 1, November 1975

INSTRUCTION CONCERNING PRENATAL RADIATION EXPOSURE

The position of Regulatory Guide is accepted.

REGULATORY GUIDE 8.14

Revision 1, August 1977

PERSONNEL NEUTRON DOSIMETERS

The position of Regulatory Guide is accepted. Refer to Subsection 12.5.2.4.4.

REGULATORY GUIDE 8.15

Revision 0, October 1976

ACCEPTABLE PROGRAM FOR RESPIRATORY PROTECTION

The position of Regulatory Guide is accepted. Refer to Section 12.5.

REGULATORY GUIDE 8.19

Revision 1, June 1979

OCCUPATIONAL RADIATION DOSE ASSESSMENT IN
LIGHT-WATER REACTOR POWER PLANTS
DESIGN STAGE MAN-REM ESTIMATES

The YGN 3&4 complies with the regulatory position of this Regulatory Guide. Refer to Section 12.4 for further information.

APPENDIX 1B - SUMMARY OF TMI-2 ACTION ITEM RESPONSES

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APPENDIX 1B - SUMMARY OF TMI-2 ACTION ITEM RESPONSES

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1B.1 INTRODUCTION

The accident at Three Mile Island Unit 2 (TMI-2) resulted in requirements that were developed from the recommendations of several groups established to investigate the accident. These groups include Congress, the General Accounting Office, the President's Commission on the Accident at Three Mile Island, the NRC Special Inquiry Group, the NRC Advisory Committee on Reactor Safeguards, the Lessons Learned Task Force, the Bulletins and Orders Task Force of the NRC Office of Nuclear Reactor Regulation, the Special Review Group of the NRC Office of Inspection and Enforcement, the NRC Staff Siting Task Force and Emergency Preparedness Task Force, and the NRC Offices of Standards Development and Nuclear Regulatory Research. The Report NUREG-0660, entitled "NRC Action Plan Developed as a Result of the TMI-2 Accident" (referred to as Action Plan), was developed to provide a comprehensive and integrated plan for the actions now judged necessary by the NRC to correct or improve the regulation and operation of nuclear facilities. The Action Plan was based on the experience from the TMI-2 accident and the recommendations of the investigating groups.

Additionally, the NRC has identified a discrete set of licensing requirements related to TMI-2 in the Action Plan for plants that are scheduled to receive an operating license. The report NUREG-0718, entitled" Licensing Requirements for Pending Applications for Construction Permits and Manufacturing License", Rev.2, January 1982, was issued. SRP Chapter 13 was also updated after the TMI-2 accident to address operations-related TMI action items. Taken together, these documents identify the specific items from NUREG-0660 that have been approved by the Commission for implementation at nuclear power plants.

In this appendix, each requirement from NUREG-0718 and SRP Chapter 13 related to TMI-2 that applies to YGN 3&4 is addressed in sequence. For NUREG-0718, only applicable items assigned to Categories 3, 4, and 5 are discussed.

(Items from Categories 1 and 2 are not applicable to CP or OL applications. NUREG-0718, Appendix B, provides the requirements for the Category 3, 4, and 5 items.) Table 1B-1 provides a list of NUREG-0718 Appendix B items which are not applicable to YGN 3&4.

Reference to secondary USNRC guidance documents (for example, NUREG-0737) is made in the compliance discussions in cases where these secondary documents provide more detailed acceptance criteria than is contained in NUREG-0718 or the SRP.

Also, it should be noted that the Owner has incorporated design information related to many of the TMI Action Items into the main text; therefore, reference to sections in the main text is made where applicable.

1B.2 COMPLIANCE WITH TMI-2 ACTION ITEMS

I.A.1.1 Shift Technical Advisor

Action Item

Each licensee shall provide an on-shift technical advisor to the shift supervisor. The shift technical advisor (STA) may serve more than one unit at a multinuit site if qualified to perform the advisor function for the various units.

The STA shall have a bachelor's degree or equivalent in a scientific or engineering discipline and have received specific training in the response and analysis of the plant for transients and accidents. The STA shall also receive training in plant design and layout, including the capabilities of instrumentation and controls in the control room. The licensee shall assign normal duties to the STAs that pertain to the engineering aspects of assuring safe operations of the plant, including the review and evaluation of operating experience. Additional clarifications were provided in NUREG-0737.

Response

KEPCO complies with the intent of this requirement. Refer to Section 13.1.

I.A.1.3 Shift Manning

Action Item

The NRC letter of July 31, 1980 from D. G. Eisenhut to all power reactor licensees and applicants set forth the interim criteria for shift staffing. NUREG-0737 provided changes and detailed clarifications.

Response

KEPCO complies with the intent of this requirement. Refer to Section 13.1

I.A.2.1 Immediate Upgrading of Reactor Operator and Senior Reactor Operator Training and Qualifications

Action Item

The original requirement was that effective December 1, 1980, an applicant for a senior reactor operator (SRO) license will be required to have been a licensed operator for 1 year. NUREG-0737 provided changes that will permit various paths to give experience equivalent to 1 year's experience as a licensed operator.

Response

KEPCO complies with this requirement. The Enforcement Decree of the Korean Atomic Energy Act requires 2 years of experience as an RO.

I.A.2.2 Training and Qualification of Operations Personnel

Action Item

Under the TMI Action Plan, the NRC may require reactor licensees to review their training and qualification programs for all operations personnel. This is interpreted to include licensed and auxiliary operators, technicians, maintenance personnel, and supervisors. The review is to examine current practices in light of the safety significance of the duties of the operations staff. If the review determines that the current practices adequately assure proper safety-related staff conduct, then documentation of the justification for this determination is required. The documentation need not be submitted to the NRC but must be maintained on site. If the review uncovers inadequacies, the licensee is required to upgrade the training and qualification practices to ensure adequate performance of operations personnel.

Response

As stated in NUREG-0933, in June 1985, the NRC recognized that the industry had made progress in developing programs to improve nuclear utility training and personnel qualification. As a result, the Commission adopted a Policy Statement on Training and Qualifications which made the training accreditation program managed by INPO the focus of training improvement in the industry. Thus, this item was RESOLVED and no new requirements were established.

I.A.2.3 Administration of Training Programs for Licensed Operators

Action Item

Pending accreditation of training institutions, licensees and applicants for operating licenses will assure that training center and facility instructors who teach systems, integrated responses, transient, and simulator courses demonstrate senior reactor operator (SRO) qualifications and be enrolled in appropriate requalification programs.

Response

The KEPCO training program complies with the intent of this requirement.

I.A.3.1 Revise Scope and Criteria for Licensing Examinations-Simulator Exams

Action Item

Simulator Examinations will be included as part of the licensing examinations.

Response

KEPCO complies with this requirement. Refer to Section 13.2.

I.A.4.2 Long-Term Training Simulator Upgrade

Action Item

Applicants shall describe their program for providing simulator capability for their plants. In addition, they shall describe how they will assure that their proposed simulator will correctly model their control room. Applicants shall provide sufficient information to permit the NRC staff to verify that they will have the necessary simulator capability to carry out the actions described in this Action Plan item as well as Action Plan Item II.K.3.54. Applicants shall submit, prior to the issuance of construction permits, a general discussion of how the requirements will be met. Sufficient details shall be presented to provide reasonable assurance that the requirements will be implemented properly prior to the issuance of operating licenses.

Response

As noted in NUREG-0933, "A Prioritization of Generic Safety Issues," Supplement 13, June 30, 1991, all aspects for this item were resolved with the publication of Revision 0 (April 1981) and Revision 1 (April 1987) to Regulatory Guide 1.149 and with the publication of a rule, 10 CFR 55.45, by the USNRC.

KEPCO complies with the intent of Regulatory Guide 1.149.

I.B.1.2 Evaluation of Organization and Management Improvements of Near-Term Operating License Applicants

Action Item

This NUREG-0660 item required the staff to evaluate organization and management capabilities of NTOL applicants before license issuance. NRR was to provide draft criteria and OIE was to manage an inter-office review team. The findings of the team was to be factored into the SER for each NTOL facility.

Between January 1980 and July 1980, 6 NTOLs (Sequoyah, North Anna 2, Salem 2, Diablo Canyon, McGuire, and Farley 2) were evaluated; Zion, Indian Point, and TMI-1 were also evaluated later. As part of its overall review responsibility, NRR was to manage similar reviews for other NTOL applicants.

Response

As stated in NUREG-0933, this item was resolved and no new requirements were established. However, YGN 3&4 will have an onsite safety review group, the plant nuclear safety committee (PNSC), similar to the type of onsite groups established in the U.S. in response to this item. The PNSC is discussed in Section 13.4 and ~~Subsection 16.6~~.
ZTS chapter3 1.6

I.C.1 Short-Term Accident Analysis and Procedure Revision

Action Item

The objective off this task is to improve the quality of procedures to provide greater assurance that operator and staff actions are technically correct, explicit, and easily understood for normal, transient, and accident conditions. The overall content, wording, and format of procedures that affect plant operation, administration, maintenance, testing, and surveillance will be included. This item consists of four parts:

I.C.1 (1) Small Break LOCA's,
I.C.1 (2) Inadequate Core Cooling,
I.C.1 (3) Transients and Accidents, and
I.C.1 (4) Confirmatory Analyses of Selected Transients.

Response

YGN 3&4 complies with items I.C.1 (1), (2), and (3). KEPCO will develop emergency operating procedures from the ABB-CE guidelines. As stated in NUREG-0933, for item I.C.1 (4), this item was RESOLVED and no new requirements were established.

I.C.2 Shift and Relief Turnover Procedures

Action Item

Shift and relief turnover is required to ensure that each oncoming shift is aware of critical plant status information and system availability prior to assuming duty.

Response

KEPCO complies with this requirement.

I.C.3 Shift Supervisor Duties

Action Item

NRC required licensees and applicants to review and revise an necessary plant procedures and directives to assure that the duties, responsibilities, and authority were properly defined to establish a definite line of command and clear delineation of the command decision authority of the supervisor in the control room relative to other plant management personnel. These letters also emphasized the primary management responsibility of the shift supervisor for safe operation of the plant. Training programs for shift supervisors were required to emphasize and reinforce the responsibility for safe operation and management function of the shift supervisor to assure safe operation of the plant.

Response

The shift supervisor duties comply with this requirement.

I.C.4 Control Room Access

Action Item

NRC letters dated September 13 and 27, October 10 and 30, and November 9, 1979, were sent to all licensees and applicants requiring that the authority and responsibilities of the person in charge of control room access and clear lines of authority and responsibility in the control room in the event of an emergency be established in conformance to item 2.2.2.a of NUREG-0578.

Response

KEPCO complies with the intent of this item by imposing restrictions on control room access.

I.C.5 Procedures for Feedback of Operating, Design and Construction Experience

Action Item

Applicants shall submit a description of their administrative procedures for evaluating operating, design, and construction experience and describe how they will assure that applicable important industry experiences originating from both within and outside the applicant's construction organization will be provided in a timely manner to those designing and constructing the plant. Applicants shall submit a general discussion of how the requirements will be met. These procedures shall: (1) Clearly identify organization responsibilities for review and identification of these important experiences and the feedback of pertinent information to those responsible for designing and constructing the plant; (2) Identify the administrative and technical review steps necessary in implementing applicable important experiences; (3) Identify the recipients provide means through which such information can be readily related to the job functions of the recipients; (4) Assure that applicant and contractor personnel do not routinely receive extraneous and unimportant experience-related information in such volume that it would obscure priority information or otherwise detract from overall job performance and proficiency; (5) Provide suitable checks to assure that conflicting or contradictory information is not conveyed to applicant and contractor personnel for implementation until resolution is reached; and (6) Provide practical interim audits to assure that the feedback program functions effectively at all levels. Sufficient detail shall be presented to provide reasonable assurance that the requirements will be implemented properly prior to the issuance of construction permits or manufacturing license.

Response

KEPCO or its designee routinely reviews, in accordance with KEPCO administrative procedures, information from a variety of sources, including: NRC Power Reactor Events, NRC Bulletins and Information Notices, and INPO/NSAC Significant Operating Experience Reports, for information that may impact plant safety. Recommendations for improvements in plant design and operation resulting from these reviews are reported to KEPCO in a timely manner.

I.C.6 Verify Correct Performance of Operating Activities

Action Item

This requires that licensees procedures be reviewed and revised, as necessary, to assure that an effective system of verifying the correct performance of operating activities is provided as a means of reducing human errors and improving the quality of normal operations. This will reduce the frequency of occurrence of situations that could result in or contribute to accidents.

Such a verification system may include automatic system status monitoring, human verification of operations, and verification of maintenance activities independent of the people performing the activity.

Response

KEPCO complies with the intent of this requirement. The YGN 3&4 design includes automatic system status monitoring in conjunction with administrative controls.

I.C.9 Long-Term Program Plan for Upgrading of Procedures

Action Item

Applicants shall describe their program plan, which is to begin during construction and follow into operation, for integrating and expanding current efforts to improve plant procedures. The scope of the program shall include emergency procedures, reliability anlysis, human factors engineering, crisis management and operator training. Applicants shall also insure that their program will be coordinated, to the extent possible, with INPO and other industry group efforts. Applicants will submit, prior to the issuance of construction permits, a general discussion of how the requirements will be met. Sufficient detail shall be presented to provide reasonable assurance that the requirements will be implemented properly prior to the issuance of operating licenses.

Response

As noted in NUREG-0933, Supplement 13, June 30, 1991, the NRC effort for this item was to develop a long-term program for upgrading plant procedures. The part of I.C.9 relating to emergency operating procedures was to be implemented in accordance with I.C.1 of NUREG-0737, Supplement 1. These requirements have now been included in Rev. 1 of SRP Section 13.5.2 and Appendix A, Rev. 0 of SRP 13.5.2, July 1985. The remaining part of I.C.9 was resolved in June 1985 by the USNRC and resulted in no new requirements.

YGN 3&4 complies with the intent of SRP 13.5.2. Refer to Subsection 13.5.2.

I.D.1 Control Room Design Reviews

Action Item

Applicants shall provide preliminary design information at a level consistent with that normally required at the construction permit stage of review. Applicants shall provide a general discussion of their approach to control room designs that reflect human factor principles by specifying the design concept selected and the supporting design bases and criteria. Cosmetic revisions to conventional (1960 technology) designs are unacceptable. Applicants shall also demonstrate that the design concept is technically feasible and within the state of the art, and that there exists reasonable assurance that the requirements will be implemented properly prior to the issuance of operating licenses. Applicants shall commit to control room designs reflecting human factors principles prior to issuance of a CP or ML (manufacturing license) and shall supply design information for review prior to committing to fabrication or revision of fabricated control room panels and layouts.

Response

The intent of this Action Item is met. Refer to Chapter 18, "Human Factors Engineering," for further information.

I.D.2 Plant Safety Parameter Display Console

Action Item

Applicants shall describe how they intend to meet the staff criteria contained in NUREG-0696 for a plant safety parameter display console. The console shall display to operators a minimum set of parameters defining the safety status of the plant, capable of displaying a full range of important plant parameters and data trends on demand, and capable of indicating when process limits are being approached or exceeded. Applicants shall, to the extent possible, provide preliminary design information at a level consistent with that normally required at the construction permit stage of review. Where new designs are involved, applicants shall provide a general discussion of their approach to meeting the requirements by specifying the design concept selected and the supporting design bases and criteria. Applicants shall also demonstrate that the design concept is technically feasible and within the state of the art, and that there exists reasonable assurance that the requirements will be implemented properly prior to the issuance of operating licenses.

Response

A plant safety parameter display console which is called the critical function monitoring system (CFMS) on YGN 3&4 is provided and is described in Subsection 7.7.1.3.4 and Section 18.2. The CFMS displays to the operator a minimum set of parameters defining the safety status of the plant. The CFMS is capable of displaying a full range of important plant parameters and data trends on demand, and indicates when process limits are being approached or exceeded.

I.D.3 Safety System Status Monitoring

Action Item

Applicants shall describe how their design conforms to Regulatory Guide 1.47, "Bypassed and Inoperable Status Indication for Nuclear Power Plant Safety Systems." Applicants shall, to the extent possible, provide preliminary design information at a level consistent with that normally required at the construction permit stage of review. Where new designs are involved, applicants shall provide a general discussion of their approach to meeting the requirements by specifying the design concept selected and the supporting design bases and criteria. Applicants shall also demonstrate that the design concept is technically feasible and within the state of the art, and that there exists reasonable assurance that the requirements will be implemented properly prior to the issuance of operating licenses.

Response

Compliance with this item is discussed in Subsection 7.1.2.19 and Section 7.5. Compliance with Regulatory Guide 1.47 is also discussed in Appendix 1A.

I.F.1 Expand QA List

Action Item

Prior to issuance of the construction permits or manufacturing license, applicants shall revise their QA programs by expanding their QA lists to include all items and activities affecting safety as defined by Regulatory Guide 1.29 and Appendix A to 10 CFR Part 50, and shall provide a commitment to apply the revised QA program to all such items and activities.

Response

This item was resolved with no new USNRC requirements issued.

I.F.2 Develop More Detailed QA Criteria

Action Item

Applicants shall describe the changes to their QA programs that have resulted from their review of the accident at TMI-2. In addition, applicants shall address the appropriate matters discussed in this Action Plan item, including the establishment of a quality assurance (QA) program based on consideration of: (a) ensuring independence of the organization performing checking functions from the organization responsible for performing the functions; (b) performing quality assurance/quality control functions at construction sites to the maximum feasible extent; (c) including QA personnel in the documented review of and concurrence in quality related procedures associated with design, construction and installation; (d) establishing criteria for determining QA programmatic requirements; (e) establishing qualification requirements for QA and QC personnel; (f) sizing the QA staff commensurate with its duties and responsibilities; (g) establishing procedures for maintenance of "as-built" documentation; and (h) providing a QA role in design and analysis activities. Applicants shall submit, prior to the issuance of the construction permits or manufacturing license, a revised description of their QA program that includes consideration of these matters.

Response

As noted in NUREG-0933, the USNRC's overall objective for this issue is the improvement of the QA program for design, construction, and operations to provide greater assurance that plant design, construction, and operational activities are conducted in a manner commensurate with their importance to safety. NUREG-0933 also states that Items I.F.2(2), I.F.2(3), I.F.2(6) and I.F.2(9) were included in the July 1981 revision of SRP Chapter 17. In the "Conclusion" for this item it is indicated that the remaining I.F.2 items are resolved by the USNRC with reprioritization as "DROP".

KEPCO's degree of conformance with SRP Chapter 17 is considered sufficient for addressing I.F.2.

II.B.1 Reactor Coolant System Vents

Action Item

Applicants shall modify their plant designs as necessary to provide the capability of high point venting of noncondensible gases from the reactor coolant system, and other systems that may be required to maintain adequate core cooling. Systems to achieve this capability shall be capable of being operated from the control room and their operation shall not lead to an unacceptable increase in the probability of loss-of-coolant accident or an unacceptable challenge to containment integrity. Applicants shall, to the extent possible, provide preliminary design information at a level consistent with that normally required at the construction permit stage of review. Where new designs are involved, applicants shall provide a general discussion of their approach to meeting these requirements by specifying the design concept selected and the supporting design bases and criteria. Applicants shall also demonstrate that the design concept is technically feasible and within the state of the art, and that there exists reasonable assurance that the requirements will be implemented properly prior to the issuance of operating licenses.

Response

Compliance is discussed in Subsection 5.4.15.

II.B.2 Plant Shielding to Provide Access to Vital Areas and Protect Safety Equipment for Postaccident Operation

Action Item

Applicants shall (1) perform radiation and shielding design reviews of spaces around systems that may, as a result of an accident, contain TID 14844 source term radioactive material and (2) implement plant design modifications necessary to permit adequate access to important areas and to protect safety equipment from the radiation environment. Applicants shall, to the extent possible, provide preliminary design information at a level consistent with that normally required at the construction permit stage of review. Where new designs are involved, applicants shall provide a general discussion of their approach to meeting the requirements by specifying the design concept selected and the supporting design bases and criteria. Applicants shall also demonstrate that the design concept is technically feasible and within the state of the art, and that there exists reasonable assurance that the requirements will be implemented properly prior to the issuance of operating licenses.

Response

YGN 3&4 complies with this action item. The plant shielding design to control access to vital areas and protect safety equipment for accident operation is discussed in Section 12.3.

II.B.3 Post-Accident Sampling

Action Item

Applicants shall (1) review the reactor coolant and containment atmosphere sampling system designs and the radiological spectrum and chemical analysis facility designs, and (2) modify their plant designs as necessary to provide a capability to promptly obtain and analyze samples from the reactor coolant system and containment that may contain TID 14844 source term radioactive materials without radiation exposures to any individual exceeding 5 rem to the whole-body or 75 rem to the extremities. Materials to be analyzed and quantified include certain radionuclides that are indicators of the degree of core damage (e.g., noble gases, iodines and cesiums, and nonvolatile isotopes), hydrogen in the contianment atmosphere, dissolved gases, chloride, and boron concentrations. Applicants shall, to the extent possible, provide preliminary design information at a level consistent with that normally required at the construction permit stage of review. Where new designs are involved, applicants shall provide a general discussion of their approach to meeting the requirements by specifying the design concept selected and the supporting design bases and criteria. Applicants shall also demonstrate that the design concept is technically feasible and within the state of the art, and that there exists reasonable assurance that the requirements will be implemented properly prior to the issuance of operating licenses.

Response

Compliance is discussed in Subsection 9.3.2.

II.B.4 Training for Mitigating Core Damage

Action Item

Licensees are required to develop a training program to teach the use of installed equipment and systems to control or mitigate accidents in which the core is severely damaged. They must then implement the training program.

Response

The KEPCO training program includes the type of training required by this action item.

II.B.8 Rulemaking Proceeding on Degraded Core Accidents

Action Item

Applicants shall:

(1) Commit to performing a site/plant-specific probabilistic risk assessment and incorporating the results of the assessment into the design of the facility. The commitment must include a program plan, acceptable to the staff, that demonstrates how the risk assessment program will be scheduled so as to influence system designs as they are being developed. The assessment shall be completed and submitted to NRC within two years of issuance of the construction permit. The outcome of this study and the NRC review of it will be a determination of specific preventive and mitigative actions to be implemented to reduce these risks.

A prevention feature that must be considered is an additional decay heat removal system whose functional requirements and criteria would be derived from the PRA study.

It is the aim of the Commission through these assessments to seek such improvements in the reliability of core and containment heat removal systems as are significant and practical and do not impact excessively on the plant. Applicants are encouraged to take steps that are in harmony with this aim.

(2) Include provisions in the containment design for one or more dedicated penetrations, equivalent in size to a single 3-foot diameter opening. This shall be done in order not to preclude the installation of systems to prevent containment failure, such as filtered vented containment systems.

(3) Provide a system for hydrogen control capable of handling hydrogen generated by the equivalent of a 100% fuel-clad metal water reaction. Preliminary design information on the tentatively preferred system option of those being evaluated in paragrah (5) of this section is sufficient at the construction permit stage. The hydrogen control system and associated systems shall provide, with reasonable assurance, that:

(a) Uniformly distributed hydrogen concentrations in the containment do not exceed 10% during and following an accident that release an equivalent amount of hydrogen as would be generated from a 100% fuel clad metal-water reaction, or that the postaccident atmosphere will not support hydrogen combustion.

(b) Combustible concentrations of hydrogen will not collect in areas where unintended combustion or detonation could cause loss of containment integrity or loss of appropriate mitigating features.

(c) Equipment necessary for achieving and maintaining safe shutdown of the plant and maintaining containment integrity will perform its safety function during and after being exposed to the environmental conditions attendant with the release of hydrogen generated by the equivalent of a 100% fuel-clad metal water reaction including the environmental conditions created by activiation of the hydrogen control system.

(d) If the method chosen for hydrogen control is a postaccident inerting system, inadvertent actuation of the system can be safely accommodated during plant operation.

(4) Provide preliminary design information at a level consistent with that normally required at the construction permit stage of review sufficient to demonstrate that:

(a) Containment integrity will be maintained (i.e., for steel containments by meeting the requirements of the ASME Boiler and Pressure Vessel Code, Division 1, Subsubarticle NE-3220, Service Level C Limits, except that evaluation of instability is not required, considering pressure and dead load alone. For concrete containments by meeting the requirements of the ASME Boiler and Pressure Vessel Code, Division 2, Subsubarticle CC-3720, Factored Load Category, considering pressure and dead load alone) during an accident that releases hydrogen generated from 100% fuel clad metal-water reaction accompanied by either hydrogen burning or the added pressure from postaccident inerting assuming carbon dioxide is the inerting agent. As a minimum, the specific code requirements set froth above appropriate for each type of containment will be met for a combination of dead load and an internal pressure of 45 psig. Modest deviations from these criteria will be considered by the staff, if good cause is shown by an applicant. Systems necessary to ensure containment integrity shall also be demonstrated to perfrom their function under these conditions.

(b) (1) Containment structure loadings produced by an inadvertent full activation of a postaccident inerting hydrogen control system (assuming carbon dioxide), but not including seismic or design basis accident loadings will not produce stresses in steel containments in excess of the limits set froth in the ASME Boiler and Pressure Vessel Code, Division 1, Subsubarticle NE-3220, Service Level A Limits, except that evaluation of instability is not required (for concrete containments the loadings specified above will not produce strains in the containment liner in excess of the limits set forth in the ASME Boiler and Pressure Vessel Code, Division 2, Subsubarticle CC-3720, Service Load Category), (2) the containment has the capability to safely withstand pressure tests at 1.10 and 1.15 times (for steel and concrete containments, respectively) the pressure calculated to result from carbon dioxide inerting.

(5) Perform an evaluation of alternative hydrogen control systems that would satisfy the requirements of paragraph (3) of this section. As a minimum, include consideration of a hydrogen ignition and postaccident inerting system. The evaluation, to be completed no later than two years following the issuance of the construction permit or the manufacturing license, shall include:

(a) A comparison of costs and benefits of the alternative systems considered.

(b) For the selected system, analyses and test data to verify compliance with the requirements of paragraph (3) of this section.

(c) For the selected system, preliminary design descriptions of equipment, function, and layout.

Response

TMI Action Item II.B.8 appears in 10 CFR 50.34(f), "Additional TMI Related Requirements" which became effective on January 15, 1982. As stated in the rule, the requirements of 10 CFR 50.34(f) apply only to construction permit and manufacturing license applications pending at the effective date of the rule. (Note: All of the applicable plants have been cancelled.) NUREG-0718, Item II.B.8 is entitled, "Rulemaking Proceeding on Degraded Core Accidents." This title reflected the intention of the USNRC to revise 10 CFR 50. As indicated in the USNRC's "Policy Statement on Severe Reactor Accidents Regarding Future Designs and Existing Plants," dated August 8, 1985, the advance notification of rulemaking for this action item has been withdrawn. This action signifies that the USNRC has no current plan to change 10 CFR 50. Although the policy statement does recommend that future plants comply with 10 CFR 50.34(f), this is not a requirement. As stated in the Federal Resister (50 FR 3218): "While the Commission's policy statement urges reactor designers to make safety improvements in the designs of future plants, it does nothing to require that improvements be made."

NUREG-0933 (Supplement dated December 31, 1985) states, in regard to Item II.B.8, that "the hydrogen control aspect of this item resulted in a Hydrogen Control Rule that was approved by the Commission and published in the Federal Register on January 25, 1985." (The rule referred to is 10 CFR 50.44 on combustible gas control systems. YGN 3&4 complies with this rule.) NUREG-0933 also states that other aspects related to severe accidents "will be dealt with for future and existing plants through procedures and ongoing severe accident programs identified in the Policy Statement" (i.e., NUREG-1070). Consequently, Item II.B.8 has been closed-out.

More recently, the USNRC issued Generic Letter 88-20 (IPE; Individual Plant Examination for Severe Accident Vulnerabilities), dated November 23, 1988, which requires each existing plant to perform a systematic examination to identify any plant-specific vulnerabilities to severe accidents. NUREG-1335 (Individual Plant Examination Submittal Guidance) provides format and content guidelines for the utility submittal. YGN 3&4 is being examined to determine its vulnerability to severe accidents in the IPE program specified in Generic

Letter 88-20 and NUREG-1335. This program will include an evaluation of external events in accordance with the guidelines of Generic Letter 88-20, Supplement 4, dated June 28, 1991.

II.D.1 Testing Requirements (RCS Safety Valves)

Action Item

Applicants and their agents shall provide a test program and associated model development and conduct tests to qualify reactor coolant system relief and safety valves and, for PWR's, PORV block valves, for all fluid conditions expected under operating conditions, transients and accidents. Consideration of anticipated transient without scram (ATWS) conditions shall be included in the test program. Actual testing under ATWS conditions need not be carried out until subsequent phases of the test program are developed. Applicants shall submit, prior to the issuance of the construction permits or manufacturing license, a general explanation of how the testing requirements will be met. Sufficient detail should be presented to provide reasonable assurance that the requirements will be implemented properly prior to the issuance of operating licenses. Applicants shall (1) demonstrate the applicability of the generic tests conducted under II.D.1 to their particular plants and (2) modify their plant designs as necessary. Applicants shall commit, prior to the issuance of the construction permits or manufacturing license, to comply with these requirements and shall submit within six months following the completion, of the generic tests or the issuance of construction permits, whichever is later, a detailed explanation of how the test results will be incorporated in the plant design. Sufficient detail should be presented to provide reasonable assurance that the requirements resulting from the test will be implemented properly prior to the issuance of operating licenses.

Response

Compliance is discussed in Subsections 5.2.2 and 5.4.13.

II.D.3 Relief and Safety Valve Position Indication

Action Item

Applicants shall modify their plant designs as necessary to provide direct indication of relief and safety valve position in the control room. Applicants shall, to the extent possible, provide preliminary design information at a level consistent with that normally required at the construction permit stage of review. Where new designs are involved, applicants shall provide a general descussion of their approach to meeting the requirements by specifying the design concept selected and the supporting design bases and criteria. Applicants shall also demonstrate that the design concept is technically feasible that the requirements will be implemented properly prior to issuance of operating licenses.

Response

Compliance is discussed in Subsections 5.2.5 and 7.7.1. An acoustic leak monitoring system (ALMS) is provided to detect leakage from each pressurizer safety valve.

II.E.1.1 Auxiliary Feedwater System Evaluation

Action Item

Applicants shall perform a reevaluation of their proposed auxiliary feedwater (AFW) system. This reevaluation shall include the following:

(1) Performance of simplified auxiliary feedwater system reliability analyses using event-tree and fault-tree logic techniques to determine the potential for AFW system failure under various loss of main feedwater transient conditions, with particular emphasis being given to determining potential failures that could result from human errors, common causes, single point vulnerabilities, and test and maintenance outages. The results of this evaluation shall be compared with the results of the NRC staff's generic AFW system evaluation published in Appendix III to NUREG-0611 and Appendix III to NUREG-0635. Applicants with plants with AFW systems with relatively low reliabilities shall submit proposed design changes and/or procedural actions which will improve the relative reliability of the AFW system to above average. Applicants whose plant designs do not include high head high pressure injection system pumps for use in the feed and bleed mode of decay heat removal in case of AFW system failure shall assure that their AFW system has a very high reliability relative to those AFW systems evaluated by the NRC and staff and reported in NUREG-0611 and NUREG-0635 respectively.
(2) Completion of a deterministic review of the AFW system using the acceptance criteria of Standard Review Plan Section 10.4.9 as principal guidance. This requirement applies to those plants where the Standard Review Plan was not used as criteria during the NRC staff's CP review.

(3) Reevaluation of the AFW system flow design bases and criteria. Applicants shall provide sufficient information to describe the nature of the studies, how they are to be conducted, the completion dates, and the program to assure that the results of such studies are factored into the final designs.

Response

An evaluation has been performed. The reliability analysis is provided in Appendix 10A.

II.E.1.2 Auxiliary Feedwater System Automatic Initiation and Flow Indication

Action Item

Applicants with PWR plants shall provide automatic and manual auxiliary feedwater (AFW) system initiation and auxiliary feedwater system flow indication in the control room. These systems shall be safety grade and meet the requirements specified in NUREG-0737. Applicants shall, to the extent possible, provide preliminary design information at a level consistent with that normally required at the construction permit stage of review. Where new designs are involved, applicants shall provide a general discussion of their approach to meeting the requirements by specifying the design concept selected and the supporting design bases and criteria. Applicants shall also demonstrate that the design concept is technically feasible and within the state-of-the-art, and that there exists reasonable assurance that the requirements will be implemented properly prior to the issuance of operating licenses.

Response

Compliance is described in Subsections 7.3.1 and 10.4.9.

II.E.3.1 Reliability of Power Supplies for Natural Circulation

Action Item

Applicants shall (1) upgrade the power supplies for the pressurizer heaters and associated motive and control power interfaces to meet the applicable requirements specified in NUREG-0737, and (2) establish procedures and training for maintaining the reactor coolant system at hot standby conditions with only onsite power available.

Applicants shall, to the extent possible, provide preliminary design information at a level consistent with that normally reguired at the construction permit stage of review. Where new designs are involved, applicants shall provide a general discussion of their approach to meeting the requirements by specifying the design concept selected and the supporting design bases and criteria. Applicants shall also demonstrate that the design concept is technically feasible and within the state-of-the-art, and that there exists reasonable assurance that the requirements will be implemented properly prior to the issuance of operating licenses.

Response

Compliance is discussed in Subsection 8.3.1.

II.E.4.1 Dedicated Penetration (Hydrogen Recombiners)

Action Item

Applicants for plant designs with external hydrogen recombiners shall modify their applications as necessry to include redundant dedicated containment pentrations so that, assuming a single failure, the recombiner systems can be connected to the containment atmosphere. Applicants shall submit, prior to the issuance of construction permits or the manufacturing license, a detailed explanation of how the requirements will be met in order to provide reasonable assurance that the requirements will be implemented properly.

Response

Compliance is discussed in Subsection 6.2.5.

II.E.4.2 Isolation Dependability

Action Item

Containment isolation system designs shall comply with the recommendations of Standard Review Plan Section 6.2.4.

All plants shall give careful consideration to the definition of essential and nonessential systems, identify each system determined to be essential, identify each system determined to be nonessential, and describe the basis for selection of each essential system. All nonessential systems shall be automatically isolated by the containment isolation signal. Revision 2 to Regulatory Guide 1.141 will contain guidance on the classification of essential versus non-essential systems and is due to be issued by June 1981.

For postaccident situations, each nonessential penetration (except instrument lines) is required to have two isolation barriers in series that meet the requirements of General Design Criteria 54, 55, 56 and 57, as clarified by Standard Review Plan, Section 6.2.4. Isolation must be performed automatically (i.e., no credit can be given for operator action). Manual valves must be sealed closed, as defined by Standard Review Plan, Section 6.2.4, to qualify as an isolation barrier. Each automatic isolation valve in a non-essential penetration must receive diverse isolation signals.

The design of control systems for automatic containment isolation valves shall be such that resetting the isolation signal will not result in the automatic reopening of containment isolation valves. Reopening of containment isolation valves shall require deliberate operator action. Administrative provisions to close all isolation valves manually before resetting the isolation signals is not an acceptable method of meeting this requirement.

Ganged reopening of containment isolation valves is not acceptable. Reopending of isolation valves must be performed on a valve-by-valve basis, or on a line-by-line basis, provided that electrical independence and other single-failure criteria continue to be satisfied.

The containment setpoint pressure that initiates containment isolation for nonessential penetrations must be reduced to the minimum compatible with normal operating conditions. The containment pressure history during normal operation for similar operating plants should be used as a basis for arriving at an appropriate minimum pressure setpoint for initiating containment isolation. The pressure setpoint selected should be far enough above the maximum observed (or expected) pressure inside containment during normal operation so that inadvertent containment isolation does not occur during normal operation from instrument drift or fluctuations due to the accuracy of the pressure sensor. A margin of 1 psi above the maximum expected containment pressure should be adequate to account for instrument error. Any proposed values greater than 1 psi will require detailed justification.

All systems that provide a path from the containment to the environs (e.g., containment purge and vent systems) must close on a safety-grade high radiation signal.

Containment purge valves that do not satisfy the operability criteria set forth in Branch Technical Position CSB 6-4 or the Staff Interim Position of October 23, 1979, must be sealed closed as defined in SRP 6.2.4, Item II. 3f during operational conditions 1, 2, 3 and 4. Furthermore, these valves must be verified to be closed at least every 31 days.

Applicants shall, to the extent possible, provide preliminary design information at a level consistent with that normally required at the construction permit stage of review. Where new designs are involved, applicants shall provide a general discussion of their approach to meeting the requirements by specifying the design concept selected and the supporting design bases and criteria.

Applicants shall also demonstrate that the design concept is technically feasible and within the state-of-the-art, and that there exists reasonable assurance that the requirements will be implemented properly prior to the issuance of operating licenses.

Response

Compliance is described in Subsections 6.2.4. and 7.3.1, Appendix 1A, and ~~Chapter 16~~.
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II.E.4.4 Purging

Action Item

Applicants shall (1) provide a capability for containment purging/venting designed to minimize purging time, consistent with ALARA principles for occupational exposure, (2) evaluate the performance of purging and venting isolation valves against accident pressure, (3) address the interim NRC guidance on valve operability, (4) adopt procedures and restrictions consistent with the revised requirements; and (5) provide and demonstrate high assurance that the purge system will be reliably isolated under accident conditions.

Applicants shall, to the extent possible, provide preliminary design information at a level consistent with that normally required at the construction permit stage of review. Where new designs are involved, applicants shall provide a general discussion of their approach to meeting the requirements by specifying the design concept selected and the supporting design bases and criteia. Applicants shall also demonstrate that the design concept is technically feasible and within the state-of-the-art, and that there exists reasonable assurance that the requirements will be implemented properly prior to the issuance of operating liceneses.

Response

As described in NUREG-0933, under item II.E.4.4(4), this item required U.S.NRC to generically evaluate the radiological consequences of containment purging of nuclear plants while in the power operation mode. It was envisioned that, as a result of this evaluation, new requirements would be needed beyond those in SRP 6.2.4 and BTP CSB 6-4. The NRC subsequently determined that this issue was a low priority item; it was then resolved without issuance of new requirements. The valve operability guidance provided in SRP Section 6.2.4 and BTP CSB 6-4, Rev. 2, dated July 1981, was considered adequate by the USNRC.

YGN 3&4 complies with the guidance in SRP Section 6.2.4 and BTP CSB 6-4.

The containment purge system is described in Section 9.4. The containment isolation design is discussed in Subsection 6.2.4.

II.F.1 Additional Accident Monitoring Instrumentation

Action Item

Applicants shall provide instrumentation to measure, record, and readout in the control room (a) containment pressure, (b) containment water level, (c) containment hydrogen concentration, (d) containment radiation intensity (high level), and (e) noble gas effluents at all potential, accident release points. The requirements for the specific monitors are listed in NUREG-0737. Applicants shall also provide for continuous sampling of radioactive iodines and particulates in gaseous effluents from all potential, accident release points, and for onsite capability to analyze and measure these samples. Applicants shall, to the extent possible, provide preliminary design at a level consistent with that normally required at the construction permit stage of review. Where new designs are involved, applicants shall provide a general discussion of their approach to meeting the requirements by specifying the design concept selected and the supporting design bases and criteria. Applicants shall also demonstrate that the design concept is technically feasible and within the state of the art, and that there exists reasonable assurance that the requirements will be implemented properly prior to the issuance of operating licenses.

Response

Compliance for this item is described in Section 7.5. The Critical Function Monitoring System (CFMS) provides monitoring and recording capability for the parameters which are used for accident monitoring as described in NUREG-0737.

II.F.2 Identification of and Recovery from Conditions Leading to Inadequate Core Cooling

Action Item

Applicants shall describe their program for developing and implementing procedures to be used by the reactor operators to detect and recover from conditions leading to inadequate core cooling.

Applicants shall provide instruments that provide in the control room an unambiguous indication of inadequate core cooling, such as primary coolant saturation meters in PWRs, and a suitable combination of signals from indicators of coolant level in the reactor vessel and incore thermocouples in PWRs and BWRs.

Applicants shall, to the extent possible, provide preliminary design information at a level consistent with that normally required at the construction permit stage of review. Where new designs are involved, applicants shall provide a general discussion of their approach to meeting the requirements by specifying the design concept selected and the supporting dsign bases and criteria. Applicants shall also demonstrate that the design concept is technically feasible and within the state of the art, and that there exists reasonable assurance that the requirements will be implemented properly prior to the issuance of operating licenses.

Response

Compliance is discussed in Subsections 7.1.2 and 7.5.1.

II.F.3 Instrumentation for Monitoring Accident Conditions (Reg. Guide 1.97)

Action Item

Applicants shall provide in their facility design instrumentation to monitor plant variables and systems during and following an accident in accordance with defined design bases and Regulatory Guide 1.97, Rev. 2, December 1980. Designs are already established for much of the instrumentation that will be required; some of the requirements, however, may involve state-of-the-art designs or designs which have yet to be developed.

Applicants shall, to the extent possible, provide preliminary design information at a level consistent with that normally required at the construction permit stage of review. Where new designs are involved, applicants shall provide a general discussion of their approach to meeting the requirements by specifying the design concept selected and the supporting design bases and criteria. Applicants shall also demonstrate that the design concept is technically feasible and within the state-of-the-art, and that there exists reasonable assurance that the requirements will be implemented properly prior to the issuance of operating licenses.

Response

Compliance is discussed in Section 7.5. Compliance with Regulatory Guide 1.97 is also discussed in Subsection 7.1.2 and Appendix 1A.

II.G.1 Power Supplies for Pressurizer Relief Valves, Block Valves, and Level Indication

Action Item

Applicants with PWR plants shall provide power supplies for the pressurizer relief valves, block valves, and pressurizer level indicators to meet the applicable requirements specified in NUREG-0737. Level indicators shall be powered from vital buses, motive and control power connections to emergency power sources shall be through devices qualified in accordance with requirements applicable to systems important to safety, and electric power shall be provided from emergency sources. Applicants with PWR plants shall, to the extent possible, provide preliminary design information at a level consistent with that normally required at the construction permit stage of review. Where new designs are involved, applicants shall provide a general discussion of their approach to meeting the requirements by specifying the design concept selected and the support design bases and criteria. Applicants shall also demonstrate that the design concept is technically feasible and within the state of the art, and that there exists reasonable assurance that the requirements will be implemented properly prior to the issuance of operating licenses.

Response

YGN 3&4 does not have pressurizer power-operated relief valves or block valves. A discussion for this item as it pertains to pressurizer level indication is provided in Subsection 7.5.1. A discussion of power supplies for vital instrumentation is provided in Section 8.3.

II.J.3.1 Organization and Staffing to Oversee Design and Construction

Action Item

Applicants shall describe their program for the management oversight of design and construction activities. Specific items to be addressed include (1) the organizational and management structure which is singularly responsible for the direction of the design and construction of the proposed plant, (2) technical resources which are directed by the utility organization, (3) details of the interaction of design and construction within the utility organization and the manner by which the utility will assure close integration of the architect engineer and nuclear steam supply vendor, (4) proposed procedures for handling the transition to operation, and (5) the degree of top level management oversight and technical control to be exercised by the utility during design and construction, including the preparation and implementation of procedures necessary to guide the effort.

Draft NUREG-0731, "Guidelines for Utility Management Structure and Technical Resources" is the keystone for similar development of guidelines for this task. Therefore, the principal applicable elements of NUREG-0731 shall be used by CP and ML applicants in addressing this task.

Applicants shall submit detailed information in order to provide reasonable assurance that the requirements will be implemented properly prior to issuance of the construction permits or manufacturing license.

Response

This item is addressed by Action Item I.B.1.1, "Organization and Management Long-Term Improvements," as noted in NUREG-0933, Supplement 6, December 31, 1986. Most of the issues for this item have been resolved with no new USNRC requirements, however, the remaining issues will be resolved in future revisions to Regulatory Guides 1.8, "Personnel Selection and Training," and 1.33, "Quality Assurance Program Requirements (Operation)."

III.A.1.2 Upgrade Licensee Emergency Support Facilities

Action Item

Applicants shall address the requirements for a Technical Support Center, Operational Support Center and the Emergency Operations Facility. Applicants shall provide preliminary design information in accordance with the functional criteria in NUREG-0696 at a level consistent with that normally required at the construction permit stage of review. Where new designs are involved, applicants shall provide a general discussion of their approach to meeting the requirements by specifying the design concept selected and the supporting design bases and criteria. Applicants shall demonstrate that the design concept is technically feasible and within the state of the art, and that there exists reasonable assurance that the requirements will be implemented properly prior to the issuance of operating licenses.

Response

The emergency support facilities consist of the technical support center (TSC), operational support center (OSC) and emergency operation facility (EOF). The emergency support facilities meet the functional requirement of NUREG-0696 and NUREG-0737, Supplement 1. The functional description of each facility is described below.

Technical Support Center (TSC)

The common shared TSC in the Unit 3 together with the satellite TSC(STSC) in Unit 4 provides plant management and technical support to plant operating personnel during emergency conditions. The TSC with the STSC relieves the reactor operators of peripheral duties and communications not directly related to reactor system manipulations, and serves to reduce congestion in the control room. The TSC with the STSC is also used to perform emergency operations facility functions until the EOF is functional.

The TSC with the STSC has facilities to support the plant management and technical personnel who will be assigned there during an emergency and is a primary onsite communications center to the control room, OSC, EOF, Emergency Response Facility (ERF) in KEPCO head office, MOST-AEO, and KINS, for the plant during an emergency. A TSC with the STSC data system is used to analyze the plant steady state and dynamic behavior prior to and throughout the course of an accident.

The location of the TSC with the STSC is within reasonable walking distance from the respective unit control room. The TSC is located in the Unit 3 access control building at el. 150'-0" and the STSC is located in the Unit 4 computer room in main control room area. The TSC workspace of approximately 7000 ft^2 (210 pyong) is sized for a minimum of 25 persons. The TSC is large enough to provide working space for personnel, space for data system equipment, space to perform repair and maintenance, access to communication equipment, access to functional displays, and storage of plant records. The general arrangement for the TSC and the STSC is provided in Section 1.2.

Since the TSC is located within the Unit 3 access control building and the STSC in Unit 4 computer room of the primary auxiliary building, are designed to withstand most adverse conditions reasonably expected during the design life of the plant including adequate capabilities for earthquake and high winds. (Section 3.8 provides further information on structural design.)

The TSC with the STSC ventilation system functions in a manner comparable to the control room ventilation system and is described in detail in Section 9.4. The habitability of the TSC with the STSC is also discussed in Section 9.4. The human factors engineering principles incorporated in the design of the TSC with STSC includes display formats, alarms, data entry features, work space, habitability, illumination, maintenance, color coding, display character size, viewing distance, functional grouping, and flashing.

The power supply for the TSC with STSC is from a non-safety-related diesel-backed power source. The backup power source provides a high degree of reliability to meet the operational unavailability requirements.

Equipment is provided to gather, store, and display data needed in the TSC with STSC to analyze plant conditions. The data system equipment performs these functions independent of actions in the control room. The data set available to the TSC with STSC data system is complete enough to permit accurate assessment of the accident without interference from control room emergency operation. The sets of Type A, B, C, D, and E variables specified in Regulatory Guide 1.97, Revision 3 are available for display and printout in the TSC with STSC. Additionally, all sensor data and calculated variables not specified in Regulatory Guide 1.97 but included in data sets for the SPDS, for the EOF, or for transmission to an offsite location are also available for display.

The TSC with STSC has a complete and up-to-date repository of plant records and procedures available to TSC with STSC personnel to aid in their technical analysis and evaluation of emergency conditions.

Operational Support Center (OSC)

The OSC is an onsite area separate from the control room and the TSC with STSC where license operations support personnel will assemble in an emergency. The OSC provides a location where plant logistic support is coordinated during and emergency. It is located in the access control building of each unit at el. 135′-0″. The OSC has direct communications with the control room and one TSC with STSC so that the personnel reporting to OSC can be assigned to duties in support of emergency operations. The OSC communication system consists of one dedicated telephone extension to the control room, telephone extension to the TSC with STSC and one dedicated dial telephone to onsite and offsite locations. Direct voice intercommunication and reliable direct radio communications are used to supplement these telephone communication links.

Emergency Operations Facility (EOF)

The EOF is a KHNP controlled and operated offsite support center. It is located in Gunseo-Myeon Yonggwang-Gun. Equipment is provided in the EOF for the acquisition, display, and evaluation of all radiological, meteorological, and plant system data pertinent to determination of offsite protective measures. 546

The location of the EOF provides optimum functional and available characteristics for carrying out KHNP's functions specified for the EOF during radiation releases. 546

The EOF is large enough to provide working space for personnel, space for EOF data system equipment, space to perform repair and maintenance, access to communication equipment, access to functional displays, and storage of plant data records.

The design of the EOF data system equipment incorporates human factors engineering principles with consideration for both operating and maintenance personnel. The human factors engineering principles incorporated in the design of the EOF include display formats, communication, data entry features, workspace, habitability, illumination, color coding, display character size, viewing distance, functional grouping, flashing, alarms, and trending.

The EOF has reliable voice communication facilities to the TSC with STSC, the control room, and local emergency authorities. The power supply to the EOF is designed with sufficient backup power sources to provide a very high degree of reliability to meet the operational unavailability goal.

Equipment is provided to gather, store, and display data needed in the EOF to analyze and exchange information on plant conditions. The EOF data system equipment performs these functions independently from actions in the control room. The EOF electrical equipment load does not degrade the capability or reliability of any safety-related power source. Circuit transients or power supply failures and fluctuations will not cause a loss of any stored data vital to the EOF functions. The total EOF data system is designed to achieve the maximum operational availability. The plant computer and the radiation monitoring system provide the EOF data system functions. The performance of the plant computer and the radiation monitoring system with their related instrumentation and equipment is included in determining the EOF data system unavailability. Use of the plant computer and the radiation monitoring system does not degrade the integrity of data supplied to the EOF or the security of the software used to process EOF data.

The EOF technical data system will receive, store, process, and display information sufficient to perform assessments of the actual and potential onsite and offsite environmental consequences of an emergency condition. Data providing information on the general condition of the plant also is available for display in the EOF for utility resource management. The EOF data set includes radiological, meteorological, and other environmental data. As a minimum EOF data set, sensor data of the Type A, B, C, D, and E variables specified in Regulatory Guide 1.97, Revision 3, and of those meteorological variables specified in proposed Revision 1 to Regulatory Guide 1.23, "Meteorological Measurements Programs in Support of Nuclear Power Plants," and in NUREG-0654, Revision 1, Appendix 2, are available for display in the EOF.

At least 2 hours of pre-event data and 12 hours of post-event data are recorded. Capacity to record at least 2 weeks of additional post-event data with reduced time resolution is provided. Archival data storage and the capability to transfer data between active memory and archival data storage without interrupting EOF data acquisition and displays is provided for all EOF data. A sufficient number of data display devices are provided in the EOF to allow all EOF personnel to perform their assigned tasks with unhindered access. Trend-information display and time-history display capability are provided in the EOF to give EOF personnel a dynamic view of plant systems, radiological status, and environmental status during an emergency.

The EOF displays are designed so that call-up manipulation and presentation of data can be easily performed. The displays are positioned to facilitate the retrieval of information by the different functional groups in the EOF. The SPDS is also displayed in the EOF. The EOF has ready access to up-to-date plant records, procedures, and emergency plans.

III.A.2 Improving Licensee Emergency Preparedness - Long Term

Action Item

Each nuclear facility shall upgrade its emergency plans to provide reasonable assurance that adequate protective measures can and will be taken in the event of a radiological emergency. Specific criteria to meet this requirement is delineated in NUREG-0654, "Criteria for Preparation and Evaluation of

Radiological Emergency Response Plans and Preparation in Support of Nuclear Power Plants."

Response

The KEPCO emergency plan is discussed in Section 13.3.

III.D.1.1 Primary Coolant Sources Outside the Containment Structure

Action Item

Applicants shall review the designs of such systems outside containment, and their provisions for leakage control and detection, overpressurization design, discharge points for waste gas venting systems, etc., with the goal of minimizing potential exposures to workers and public following an accident, and providing reasonable assurance that excessive leakage will not prevent the use of systems needed in an emergency. Applicants shall provide for leakage control and detection in the design of systems outside containment that contain (or might contain) TID 14844 source term radioactive materials following an accident, and submit a leakage control program, including an initial test program and a schedule for retesting these systems, and the actions to be taken for minimizing leakage from such systems.

In this regard, applicants shall submit, prior to the issuance of construction permits, a general discussion of their approach to minimizing leakage from such systems outside containment in sufficient detail to provide reasonable assurance that this objective will be met satisfactorily prior to issuance of operating licenses.

Response

Integrated Leak Testing

Integrated leak tests will be performed at refueling cycle intervals or less on each system or portions of systems, which could potentially contain highly radioactive liquids or gases. Station surveillances and procedures will be used to:

a. Monitor the leak testing of piping so that the appropriate lines are examined at the required intervals.

b. Direct leak test examinations such that systems are tested at approximate operating pressures or higher.

c. Align systems such that all piping tested is properly pressurized.

d. Identify lines which contain gases that require pressure decay, and/or metered makeup testing.

e. Quantify results of leakage examinations.

f. Initiate corrective action.

Leakage observed during the performance of inservice tests will be documented and a work request generated to repair leakage. Work requests of this type will be assigned a high priority and designated as an ALARA concern to initiate a review for possible modification to reduce leakage in the future.

Systems to be Tested

The following piping systems outside containment would or could contain highly radioactive liquids and gases during or following a serious transient or accident. Portions of systems which will not be included in the integrated leak tests are identified.

a. Containment Spray (CS)

The spray additive tanks and their associated piping are excluded because this portion of the system only supplies uncontaminated hydrazine to the spray additive pumps. When all hydrazine has been supplied to the pumps, check valves prevent leakage from the containment spray system into this portion of the system.

b. Shutdown Cooling (SC)

c. Safety Injection (SI)

The refueling water tank and its associated piping is not included because it provides relatively clean borated water to the HPSI pumps, containment spray pumps, and LPSI pumps during an accident. When the level in the tank reaches its low level, suction is transferred to the containment recirculation sump and the RWT becomes isolated from the system which prevents highly contaminated water from entering it.

The SIT fill lines are excluded because they are used to fill the SITs and would not be in use during an accident and could not become contaminated with highly radioactive water.

The leakoff lines from the recirculation line isolation valve encapsulation tanks have not been included because there is little potential for these lines to become highly radioactive.

d. Primary and Process Sampling (PX and PS)

The pressurizer steam and surge line sample lines, reactor coolant sample line, and shutdown cooling sample lines are included. Also, the sample lines from the reactor drain tank, ESF pump room sumps, the SI recirculation line, and the containment air sample line are included. These sample lines will be tested up to the sample panels.

The remainder of the PX system and the entire PS system will be excluded. The systems are normally isolated and sampling is intermittent. In addition, all piping is 1 inch or less with most piping in the 3/8-inch to 3/4-inch range. Piping lengths are minimized for sampling considerations. The sample panel areas, where the greatest probability of leakage exists, are kept at a negative pressure. Also, the system is frequently operated by the Chemistry Section personnel during normal plant operation. Considering all the above system characteristics, it would be unlikely that a loss of flow would go undetected.

e. Radioactive Drains (DE)

The casing drain lines from the following pumps are included:

1. HPSI pumps
2. LPSI pumps
3. containment spray pumps

The remainder of the DE system is excluded.

Integrated Test Leak Acceptance Criteria

After Unit 3 reaches full power operation, KEPCO will submit a report of all recorded leakage and all preventive maintenance performed as the direct result of the evaluation of this leakage. The report will also identify general leakage criteria to be applied during the first fuel cycle as the basis for instituting corrective action in the form of preventive maintenance. Through this program's commitment to generate work requests for all practically repairable leakage problems, levels of leakage will be kept as low as practicable. Because leakage problems presenting ALARA concerns will be reviewed, the leakage criteria can be refined over time as more information is accumulated through testing. Thus, the criteria can be revised to incorporate new modifications and techniques designed to keep leakage as low as practicable. In other words, the leakage criteria will be designed such that it excludes all practically preventable leakage, based upon current design, repair, and operating techniques.

Prior to the start of the second fuel cycle, the general criteria will be revised based on the experience gained during the first operating cycle on Unit 3. These revised criteria will be used as the basis for the long term leakage monitoring program on Units 3 and 4.

The initial system leak monitoring data will be taken after fuel load, during the startup testing. Leakage rates observed during this period provide a better baseline than those taken prior to fuel loading. Leak rates observed during preoperational testing are not necessarily representative of operating leak rates because of continuing adjustments to valve packing and seals, valve seat lapping, and the opening and closing of various mechanical joints.

Implementation of the program described above will assure that initial leak monitoring will accurately indicate leak rates under actual operating conditions.

Other Leak Testing

In addition to this integrated leak test program, all Class 1, 2, and 3 systems will be leak tested at prescribed intervals, in accordance with the requirements of Section XI of the ASME Boiler and Pressure Vessel Code, "Rules for Inservice Inspection (ISI) of Nuclear Power Plant Components," as described in Sections 5.2 and 6.6. Therefore, Class 1, 2. and 3 portions of systems excluded from this leakage program, will be leak tested through the ISI Program.

The piping and components which make up the containment penetrations are tested periodically as part of the 10 CFR 50, Appendix J leakage testing program for Type A and Type B testing (Type C testing is in accordance with the Technical Specifications).

Prior to fuel load, all systems or portions of systems constructed in accordance with ASME Section III are hydrostatically tested to 125% of the system design pressure. In the case of gaseous systems, a pneumatic type pressure decay test at 125% of system design pressure is performed. All systems in this program are tested prior to initial plant startup via the Pre-Operational Test Program. During these tests, system walkdowns are conducted by the system test engineer and deficiencies are documented for leaking and defective components. In addition to the individual system tests, integrated type tests such as integrated hot functional and emergency core cooling full flow tests are conducted. During these integrated tests, additional system walkdowns are conducted for vibrational testing and inspection of piping thermal expansion. Deficiencies are documented during these walkdowns also.

III.D.3.3 In-Plant Radiation Monitoring

Action Item

Applicants shall review their designs to ensure that provisions for monitoring inplant radiation and airborne radioactivity are appropriate for a broad range of routine and accident conditions. Applicants shall, to the extent possible, provide preliminary design information at a level consistent with that normally required at the construction permit stage of review. Where new designs are involved, applicants shall provide a general discussion of their approach to meeting the requirements by specifying the design concept selected and the supporting design bases and criteria. Applicants shall also demonstrate that the design concept is technically feasible and within the state of the art, and that there exists reasonable assurance that the requirements will be implemented properly prior to the issuance of operating licenses.

Response

Compliance is discussed in Sections 11.5 and 12.3.

III.D.3.4 Control Room Habitability

Action Item

Applicants shall review the design of their facilities for confromance to requirements stated in the Action Plan. Applicants shall evaluate potential pathways for radioactivity and radiation that may lead to control room habitability problems under accident conditions resulting in a TID 14844 source term release and make necessary design provisions to preclude such problems.

Applicants shall address prior to the issuance of the construction permits or manufacturing license, how they will implement the existing requirements set forth in this Action Plan item. Applicants shall also address the extent to which improvements have been made to prevent control room contamination via pathways not previously considered. Applicants shall, to the extent possible, provide preliminary design information at a level consistent with that normally required at the construction permit stage of review. Where new designs are involved, applicants shall provide a general discussion of their approach to meeting the requirments by specifying the design concept selected and the supporting design bases and criteria. Applicants shall also demonstrate that the design concept is technically feasible and within the state of the art, and that there exists reasonable assurance that the requirements will be implemented properly prior to the issuance of operating licenses.

Response

Compliance is discussed in Section 6.4.

TABLE 1B-1 (Sh. 1 of 2)

NUREG-0718 APPENDIX B ACTION ITEMS
NOT APPLICABLE TO YGN 3&4

ACTION ITEM	TITLE	REMARKS
II.E.5.1	Design Evaluation	B&W* only
II.K.1.22	Describe Automatic and Manual Actions for Proper Functioning of Auxiliary Heat Removal Systems When FW System Not Operable	BWR only
II.K.2.9	Analysis and Upgrading of Integrated Control System	B&W only
II.K.2.10	Hard-Wired Safety-Grade Anticipatory Reactor Trips	B&W only
II.K.2.16	Impact of RCP Seal Damage Following Small Break LOCA	B&W only
II.K.3.2	Report on Overall Safety Effect of PORV Isolation System	Not Applicable to NSSS design for YGN 3&4. There are no pressurizer PORVs.
II.K.3.13	Separation of HPCI and RCIC System Initiation Levels - Analysis and Implementation	BWR only
II.K.3.16	Reduction of Challenges and Failures of Relief Valves - Feasibility Study and System Modification	BWR only
II.K.3.18	Modification of ADS Logic - Feasibility Study and Modification for Increased Diversity for Some Event Sequences	BWR only

*Applies to reactors manufactured by Babcock & Wilcox only.

TABLE 1B-1 (Sh. 2 of 2)

ACTION ITEM	TITLE	REMARKS
II.K.3.21	Restart of Core Spray and LPCI Systems on Low Level - Design and Modification	BWR only
II.K.3.23	Central Water Level Recording	BWR only
II.K.3.24	Confirm Adequacy of Space Cooling for HPCI and RCIC Systems	BWR only
II.K.3.28	Verify Qualification of Accumulators on ADS Valves	BWR only
II.K.3.45	Evaluate Depressurization With Other Than Full ADS	BWR only

APPENDIX 1C - IMPROVEMENT ACTION ITEMS POST FUKUSHIMA DAIICHI ACCIDENT

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1C.1 INTRODUCTION

This chapter describes implementation for the improvement action items issued in the domestic NPP safety review report performed as part of countermeasures post the Fukushima Daiichi accident.

1C.2 IMPLEMENTATION FOR ACTION ITEMS POST FUKUSHIMA DAIICHI ACCIDENT

(3-5-2) INSTALLATION OF EMERGENCY MAKEUP WATER SUPPLY LINE FOR SPENT FUEL POOL COOLING

Improvement Action Item

Provide makeup water by using fire engine and so on, and install a connector for the makeup water supply in order to obtain means for alternative heat removal under the loss of function of pumps and heat exchangers in the spent fuel pool cooling system.

Implementation of Action Item

In compliance with the action item, the external makeup water supply system was designed in the YONGGWANG Nuclear Power Plant Unit No. 3, 4, as shown in Fig. 9.I-3, so that the alternative heat removal function could be secured under the loss of function of pumps and heat exchangers in the spent fuel pool cooling system.

(3-6) MEASURES TO PREVENT FLOODING OF FINAL HEAT REMOVAL EQUIPMENT AND ESTABLISHMENT OF RECOVERY PLANS

Improvement Action Item

The spare parts required for the maintenance of Essential Service Water System Pump Motors in case of large storms and tsunamis shall be prepared and the recovery plans for the functional failure of the motors shall be established. 667

Implementation of Action Item

In compliance with the action item, the spare parts of Essential Service Water System Pump Motors have been prepared and the recovery plans for the functional failure of the motors have also been estabilshed for YONGGWANG Nuclear Plant Unit 3 & 4.

(3-A1) APPLICATION OF SAFETY-RELATED TEMPERATURE INSTRUMENT FOR SPENT FUEL POOL

Improvement Action Item

Change temperature instrument for spent fuel pool to Safety-related grade. 673

Implementation of Action Item

Temperature instrument for spent fuel pool could be improved from non safety-related to safety-related. It was designed to be observed in Main Control Room(MCR).

(3-10-1) INSTALLATION OF EMERGENCY MAKEUP WATER SUPPLY PROVISION FOR FIRE ENGINE

Improvement Action Item

Install the emergency water supply provision for supplying the emergency makeup water to fire engine, even though the fire water is not available because of a large tsunami.

691

Implementation of Action Item

in Compliance with the action item, the emergency water supply provision was designed in the YOUNGGWANG Nuclear Power Plant Unit No.3 & No.4 as shown in the system diagram Fig. 9.2-10, so that is could be secured to supply the emergency water to fire engine, even though the fire water is not available because of a large tsunami.

(3-1) SECUREMENT OF MOBILE GENERATOR

Improvement Action Item

4.16 kV mobile gas turbine generator shall be obtained at each site. 706

Implementation of Action Item

4.16 kV mobile gas turbine generator which is mounted on truck or trailer shall be obtained at each site.

4.16 kV mobile gas turbine generator, in the event of SBO and condition of unavailability of AAC D/G power occurs at the same time under long-term, shall supply the power for the essential loads of the plant using power cable long enough in length.
4.16 kV mobile gas turbine generator continuous operational rating is 3,200 kW. Power connection points are as follows: 706

- Temporary emergency power is connected in the bus, 812-EAP-01SA or 812-EAP-01SB.

(5-6) REINFORCEMENT OF POWER SUPPLY SYSTEM FOR SPDS (SAFETY PARAMETER DISPLAY SYSTEM)

Improvement ACtion Item

120/208V transportable diesel generator shall be obtained at each site.

Implementation of Action Item

120/208V transportable diesel generator which is mounted on trailer shall be obtained at each site. 739
120/208V transportable diesel generator shall supply the power for SPDS(Safety Parameter Display System) in case power loss of SPDS.
The emergency power shall be 3 phase / 120/208V / 60Hz / 30kVA rating capacity.

Power connection points are as follows:
- Temporary emergency power is connected in the bus, 120/208V PMS INSTRUMENT PANEL (3/4-815-EIP-03FN).

(4-3) INSTALLATION OF EXTERNAL INJECTION LINE FOR REACTOR EMERGENCY COOLING WATER

Improvement Action Item

753

Provide emergency cooling water by using fire engine and so on, and install a connector for emergency cooling water supply in order to obtain means for alternative cooling water under the loss of cooling water in the reactor coolant system and steam generators secondary side.

Implementation of Action Item

In compliance with the action item, the external injection systems was designed in YONGGWANG Nuclear Power Plant Unit No. 3&4 so that the alternative cooling water supply could be secured under the loss of cooling water in the reactor coolant system and steam generators secondary side.

Design Condition

external injection system	Primary side	Secondary side	
a. connect point	Reactor coolant system	steam generator	
b. Minimum flow injection	1,230 LPM @ 13.9kg/㎠.g	779 LPM @ 2.1kg/㎠.g	
c. Design basis	Through the maintenance of the structural and functional safety, prevent core melting down accident and minimize the unpredictable situations.	Through the steam generator water level recover in any intact steam generator, dissipate core residual heat from the reactor coolant system and reduce possibility of radiation leakage accidents from the reactor.	753
d. Related document	Conceptual design report, performance appraisal report in the design change document (영이34-441-1671)	Conceptual design report, performance appraisal report in the design change document (영이34-441-1671)	

(2-2-1) INSTALLATION OF WATER-TIGHT DOORS

Improvement Action Item

Installing water-tight doors in structures that are seismically designed to cope with the possibility of flooding of the emergency power sources and main safety facilities. 770

Implementation of Action Item

In complance with the action item, water-tight doors were designed in the YONGGWANG Nuclear Power Plant Unit No. 3, 4, so that the major safety function could be secured under the flooding by a TSUNAMI accident.