Status Report - HPR1000 (China)

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1. Overview

In Operation

Number of Units	Country	
8	China	Link to PRIS

Under Construction

Number of Units	Country	
19	China	Link to PRIS

Described in

Link to PRIS	--

New Concept

Number of Units	--
Year	--

Reference Plant

Name of Site	ZHANGHZHOU-1 (Link to PRIS)
Net Power Output (MWe)	1126
Gross Power Output (MWe)	1212

1.1 Introduction

HPR1000 is an evolutionary pressurized water reactor (PWR) developed by China National Nuclear Corporation (CNNC) and China General Nuclear Power Corporation (CGN). The design is based on design and operational experience from pressurized water reactors in China and incorporates design features to meet utility requirements and applicable nuclear safety requirements, including the safety issues relevant to Fukushima accident.

CNNC has developed pressurized water reactor technology based on experience gained from the design, construction, commissioning and operation of NPPs in China. The PWR Research and

Development (R&D) program of CNNC went through three phases, which are respectively represented by three successively-developed PWR models known as CNP1000, CP1000 and ACP1000. Phase I can be traced back to 1999, with the launch of research program of CNP1000. During Phase II, from 2007 to 2010, CP1000 was developed on the foundation of CNP1000 by introducing 22 major technical improvements. The preliminary design passed the peer review organized by China Nuclear Energy Association (CNEA) and PSAR has been submitted to China National Nuclear Safety Administration (NNSA) by 2010. However, the licensing of all the new NPP projects was suspended after Fukushima accident, which brought forth Phase III dominated by the R&D of ACP1000. The concept of ACP1000 was indeed proposed as a further upgraded version of CP1000 before Fukushima accident, and the key technology research program of ACP1000 was launched as early as January, 2010. After Fukushima accident, the program was gathering pace to develop a PWR incorporating additional safety features and lessons learned from the Fukushima Daiichi accident.

CGN also started an R&D program of advanced PWR after Fukushima accident, which can be divided into two steps. The first step is to develop ACPR1000 with evolutionary improvements on the basis of mature CPR1000. For the second step, ACPR1000+ was developed as an advanced PWR solution.

In April, 2013, under the coordination of the Chinese government, CNNC and CGN determined to jointly develop a model of advanced PWR based on ACP1000 and ACPR1000+, which are officially named as HPR1000 (also known as ACP1000, HL-1 or Hualong-1 before this formal brand is settled).

In August, 2014, the technical scheme of HPR1000 was reviewed by the experts organized by China National Energy Administration (NEA) and NNSA. On April 15, 2015, the State Council of China approved the construction of the HPR1000 pilot project, Fuqing NPP Unit 5&6, which located in Fujian Province, in the southeast of China. The first concrete was poured on May 7, 2015, after the safety review of PSAR was finished and Construction License was granted. This is a milestone for China to restart the nuclear power projects after Fukushima accident.

In China, the HPR1000 nuclear power technology has achieved batch construction and operation, and has been deployed in tens of plant sites such as Zhangzhou in Fujian, Fangchenggang in Guangxi, Changjiang in Hainan, Taipingling in Guangdong, Ningde in Fujian, San'ao and Jinqimen in Zhejiang, and Zhaoyuan in Shandong.

While advancing domestic engineering projects, China National Nuclear Corporation is promoting the implementation of overseas HPR1000 projects, including the operating Karachi NPP K2 and K3 units in Pakistan, as well as the under-construction Chashma Unit 5.

As of May 30, 2026, there are a total of 41 HPR1000 units in operation, under construction, and approved worldwide, making it the third-generation nuclear power technology with the largest fleet in the world. Among them, 10 units have been put into operation; 20 units are under construction; and 11 units have been approved.

1.2 Development Milestones

2015	FCD for the HPR1000 pilot project, Fuqing NPP Unit 5&6 in China, and FCD for the Karachi NPP Unit K2&K3 in Pakistan

2021	Commercial Operation for Fuqing NPP Unit 5 (China) and Karachi NPP Unit K2 (Pakistan)
2022	Commercial Operation for Fuqing NPP Unit 6 (China) and Karachi NPP Unit K3 (Pakistan)
2025	Total reported fleet: 41 units operating, under construction, or approved worldwide.

1.3 Design organization or vendor company

CNNC, CGN

1.4 Links to designer/vendor website

https://www.cnnc.com.cn/, http://www.cgnpc.com.cn/

1.5 Design Description

The reactor core of HPR1000 is loaded with 177 fuel assemblies, ensuring sufficient thermal margin while increasing output power. The design uses the CF3 fuel assembly developed by CNNC. With innovative Zircaloy cladding material and design of grids, nozzles and guide tubes, the CF3 fuel assembly is designed for the fuel management strategy and long refuelling intervals specified for HPR1000.

The application of active and passive safety design concept is another important feature for HPR1000. The design inherits the mature and reliable active technology, and also introduces passive systems as the backup for active systems in case of loss of AC power. Specifically speaking, both active and passive features are employed to guarantee the safety functions of emergency core cooling, residual heat removal, In-Vessel Retention (IVR) of the molten core, and containment heat removal. Extensive tests were performed to verify that the passive features will achieve the designed functions.

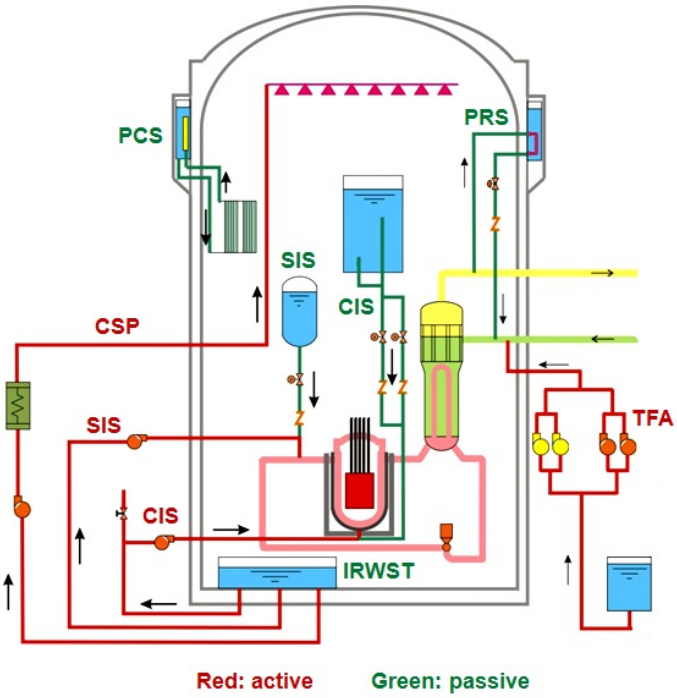


Figure 1 Safety Configuration of HPR1000 for Fuqing NPP Unit 5&6

PCS: Passive Containment heat removal System

PRS: Passive Residual heat removal System

SIS: Safety Injection System
CIS: Cavity Injection and cooling System
CSP: Containment Spray system
IRWST: In-containment Refueling Water Storage Tanks
TFA: Auxiliary feedwater system

The design of HPR1000 incorporates comprehensive severe accident prevention and mitigation measures against various severe accident phenomena, including high-pressure molten corium ejection, hydrogen detonation, basement melt-through, long-term containment overpressure and station blackout. Probabilistic Safety Assessment (PSA) results indicate a Core Damage Frequency (CDF) of less than 1.0E-6/reactor·year, and a Large Release Frequency (LRF) of less than 1.0E-7/reactor·year. The values are one order of magnitude lower than the requirements specified in Utility Requirement Document (URD) and European Utility Requirement (EUR), and meet the safety targets set by Chinese authorities for the new NPPs.

As a new model available after Fukushima accident, necessary provisions for coping with the Fukushima accident scenario have been incorporated into HPR1000, such as the emergency power and cooling water supply solutions, the improved monitoring and cooling capability of spent fuel pool, the enhanced habitability and availability of emergency facilities and so on.

The operational performance and economic goals of HPR1000 are in line with the requirements contained in utility requirements documents, e.g. plant availability, design lifetime, refueling cycle, etc. Most of the main equipment is proven and can be manufactured in China, which supports equipment procurement from suppliers available in China. Advanced features such as Leakage-Before-Break (LBB) technology, Integrated Reactor Pressure Vessel (RPV) Head Structure, etc., also reduce the cost and period of construction, maintenance and refueling.

1.6 Licensing Application Support Documents

1.7 Reactor Unit in PRIS

China	Fuqing	2 units operating
China	Fangchenggang	2 units operating
Pakistan	Karachi	2 units operating
China	Zhangzhou	2 units operating, 2 units under construction
China	Sanao	1 unit operating, 2 unit under construction
China	Shidaowan	2 units under construction
China	Jinqimen	2 units under construction
China	Ningde	2 units under construction
China	Taipingling	1 unit operating, 3 units under construction

China	Lufeng	2 units under construction
China	Changjiang	2 units under construction
Pakistan	Chashma	1 unit under construction
China	Xuwei	1 unit under construction
China	Zhaoyuan	1 unit under construction

2. Category Fields

Table 1: ARIS Category Fields

ARIS Category	Value	Units of Examples
Current/Intended Purpose	Commercial – Electric/Non-Electric	e.g. Commercial – Electric/Non-electric, Prototype/FOAK, Demonstration, Experimental
Main Intended Application (once commercial)	Baseload, Dispatchable, Non-electric (Industrial Steam Supply; District heating)	e.g. Baseload, Dispatchable, Off-grid/Remote, Mobile/Propulsion, Non-electric (specify)
Reference Location	On Coast, Inland	e.g. On Coast, Inland, Below-Ground, Floating-Fixed, Marine-Mobile, Submerged-Fixed (Other-specify)
Reference Site Design (reactor units per site)	Single Unit	e.g. Single Unit, Dual Unit, Multiple Unit (# units)
Core Coolant	H2O	H2O, D2O, He, CO2, Na, Pb, PbBi, Molten Salts, (Other-specify)
Reactor Core Size (1 core)	Thermal power > 3000 MWth	e.g. Thermal Power < 30 MWth, 30 MWth ≤ Thermal power < 1000 MWth, 1000 MWth ≤ Thermal power ≤ 3000 MWth, Thermal power > 3000 MWth
Reactor Type	PWR	e.g. PWR, BWR, HWR, SCWR, GCR, GFR, SFR, LFR, MSR, ADS
Reactor Size	Large	e.g. PWR, SMR
Neutron Moderator	H2O	e.g. H2O, D2O, He, CO2, Na, Pb, PbBi, Molten Salts, (Other-specify)

NSSS Layout	Loop-type (3 loop(s))	e.g. Loop-type (# loops), Direct-cycle, Semi-integral, Integral, Pool-type
Primary Circulation	Forced (3 pumps)	e.g. Forced (# pumps), Natural
Thermodynamic Cycle	Rankine	e.g. Rankine, Brayton, Combined-Cycle (direct/indirect)
Secondary Side Fluid	H2O	e.g. H2O, He, CO2, Na, Pb, PbBi, Molten Salts, (Other-specify)
Fuel Form	Fuel Assembly/Bundle	e.g. Fuel Assembly/Bundle, Coated Sphere, Plate, Prismatic, Contained Liquid, Liquid Fuel/Coolant
Fuel Lattice Shape	Square	e.g. Square, Hexagonal, Triangular, Cylindrical, Spherical, Other, n/a
Rods/Pins per Fuel Assembly/Bundle	17×17	e.g. #, n/a
Fuel Material Type	Oxide	e.g. Oxide, Nitride, Carbide, Metal, Molten Salt, (Other-specify)
Design Status	Operational	e.g. Conceptual, Detailed, Final (with secure suppliers)
Licensing Status	In Operation	e.g. DCR, GDR, PSAR, FSAR, Design Licensed (in Country), Under Construction (# units), In Operation (# units)
Neutron Spectrum	Thermal	e.g. Fast, Thermal

3. Parameter Fields

Table 2: ARIS Parameter Fields

ARIS Parameter	Value	Units of Examples
Plant Infrastructure		
Design Life	60	Years
Lifetime Capacity Factor	≥90	%, defined as Lifetime MWe-yrs delivered / (MWe capacity * Design Life), incl. outages
Major Planned Outages		# days every # months (specify purpose, including refuelling)

Operation / Maintenance Human Resources		# Staff in Operation / Maintenance Crew during Normal Operation
Reference Site Design		n Units/Modules
Capacity to Electric Grid	~1200 (depend on site conditions)	MWe (net to grid)
Non-electric Capacity		e.g. MW(th) heat at x ºC, m3/day desalinated water, kg/day hydrogen, etc.
In-House Plant Consumption	~86	MWe
Plant Footprint		m2 (rectangular building envelope)
Site Footprint		m2 (fenced area)
Emergency Planning Zone	7~10km	km (radius)
Releases during Normal Operation		TBq/yr (Noble Gases / Tritium Gas / Liquids)
Load Following Range and Speed	15%-100%, 5%Pn/min	x – 100%, % per minute
Seismic Design (SSE)	0.3	g (Safe-Shutdown Earthquake)
NSSS Operating Pressure (primary side)	15.5	MPa(abs), i.e. MPa(g)+0.1, at core outlet
NSSS Operating Pressure (secondary side)	6.8	MPa(abs), i.e. MPa(g)+0.1, at secondary side outlet
Primary Coolant Inventory (incl. pressurizer)		kg
Nominal Coolant Flow Rate (primary)		kg/s
Nominal Coolant Flow Rate (secondary)	1788.0	kg/s
Core Inlet Coolant Temperature	292.1	ºC
Core Outlet Coolant Temperature	327.9	ºC
Available Tempeature as Process Heat Source		ºC
NSSS Largest Component	RPV	e.g. RPV (empty), SG, Core Module (empty/fuelled), etc.

- NSSS Largest Component Length	12567	mm
- NSSS Largest Component Diameter	4340 (Inner)	mm
- NSSS Largest Component Thickness		mm
- NSSS Largest Component Weight		tonne
Reactor Vessel Material	16MND5	e.g. SS304, SS316, SA508, 800H, Hastelloy N
Steam Generator Design	Vertical, U-Tube	e.g. Vertical/Horizontal, U-Tube/ Straight/Helical, cross/counter flow
Secondary Coolant Inventory		kg
Pressurizer Design	separate vessel, steam pressurised	e.g. separate vessel, integral, steam or gas pressurised, etc.
Pressurizer Volume	53	m3 / m3 (total / liquid)
Containment Type	Concrete	Dry (single/double), Dry/Wet Well, Inerted, etc.
Containment Volume		m3
Spent Fuel Pool Capacity and Total Volume		Years of full-power operation / m3
Non Electric Application	Industrial Steam Supply	-
Core Delta Temperature		ºC
Fuel/Core		
Single Core Thermal Power	3180	MW(th)
Refuelling Cycle	18	months or “continuous”
Fuel Material	UO2	e.g. UO2, MOX, UF4, UCO
Enrichment (avg./max.)		%
Average Neutron Energy		eV
Fuel Cladding Material	ZIRLO	e.g. Zr-4, SS, TRISO, E-110, none
Number of Fuel “Units”	177	Specify as Assembly, Bundle, Plate, Sphere, or n/a
Weight of one Fuel Unit		kg

Total Fissile Loading (initial)		kg fissile material (specify isotopic and chemical composition)
% of fuel outside core during normal operation		Applicable to online refuelling and molten salt reactors
Fraction of fresh-fuel fissile material used up at discharge		%
Core Discharge Burnup		MWd/kgHM (heavy metal, eg U, Pu, Th)
Pin Burnup (max.)		MWd/kgHM
Breeding/Conversion Ratio		Fraction of fissile material bred in-situ over one fuel cycle or at equilibrium core
Reprocessing		e.g. None, Batch, Continuous (FP polishing/actinide removal), etc.
Main Reactivity Control	Rods supported by boron solution	e.g. Rods, Boron Solution, Fuel Load, Temperature, Flow Rate, Reflectors
Solid Burnable Absorber	Gd2O3	e.g. Gd2O3
Core Volume (active)		m3 (used to calculate power density)
Fast Neutron Flux at Core Pressure Boundary		n/(cm²*s)
Max. Fast Neutron Flux		n/(cm²*s)
Core Control Rod Absorber		-
Core Soluble Neutron Absorber		-
Safety Systems		
Number of Safety Trains	Active / Passive	% capacity of each train to fulfil safety function
- reactor shutdown	Passive	- / 100 / -
- core injection	Active	100 / - / 100
- decay heat removal	Active	100 /- / 100
- containment isolation and cooling	Active	100 / - / 100
- emergency AC supply (e.g. diesels)	Active	100 / - / 100

DC Power Capacity (e.g. batteries)	72	Hours
Events in which Immediate Operator Action is required	None	e.g. any internal/external initiating events, none
Limiting (shortest) Subsequent Operator Action Time	0.5	Hours (that are assumed when following EOPs)
Severe Accident Core Provisions	IVR	e.g. no core melt, IVMR, Core Catcher, Core Dump Tank, MCCI
Core Damage Frequency (CDF)	< 1.0E-6	x/reactor-year (based on reference site and location)
Severe Accident Containment Provisions	H2 combinations, Filtration and Exhaust, etc.	e.g., H2 ignitors, PARs, filtered venting etc.
Large Release Frequency (LRF)	< 1.0E-7	x/reactor-year (based on reference site and location)
Overall Build Project Costs Estimate or Range (excluding Licensing, based on the Reference Design Site and Location)		
Construction Time (nth of a kind)	62~69	Months from first concrete to criticality
Design, Project Mgmt. and Procurement Effort		Person-years (PY) [DP&P]
Construction and Commissioning Effort		PY [C&C]
Material and Equipment Overnight Capital Cost		Million US$(2015) [M&E], if built in USA
Cost Breakdown	%[C&C]/%[M&E]	
- Site Development before first concrete		(e.g. 25/10) (30/40) (20/25) (20/10) (5/15) (----------) (to add up to 100/100)
- Nuclear Island (NSSS)		
- Conventional Island (Turbine and Cooling)		
- Balance of Plant (BOP)		
- Commissioning and First Fuel Loading		
Factory / On-Site split in [C&C] effort		%/% of total [C&C] effort in PY (e.g., 60/40)
Nuclear System		
System Steam Flow Rate	1773	kg/s

System Steam Pressure	6.45	MPa
System Steam Temperature	280.3	ºC
System Feedwater Flow Rate	1788.3	kg/s
System Feedwater Temperature	226	ºC
Reactor Core		
Active Core Height	3.658	m
Equivalent Core Diameter	3.228	m
Average Linear Heat Rate	18.12	kW/m
Average Fuel Power Density		kW/kgU
Average Core Power Density		MW/m3
Outer Diameter Of Fuel Rods	9.5	mm

4. Plant Layout, Site Environment and Grid Integration

SUMMARY FOR BOOKLET

The HPR1000 adopts single-unit layout , better realizes physical isolation, effectively reduces common-mode failures of safety systems caused by events such as fires and flooding, facilitates power plant construction, operation, and maintenance, and improves the flexibility of site plan selection.

The HPR1000 design adheres to the principle of waste minimization, employing advanced radioactive waste treatment processes. The discharge of gaseous and liquid effluents meets the requirements of China's national standards (GB6249 and GB14587). The estimated annual production of solid waste packages per unit is no more than 50 m3.

The HPR1000 power plant has the capability for load-following operation. Its flexible power output adjustment function can achieve daily load following within a load scope of 15% to 100% of nominal power (Pn).

Located on the southeast coastal bay of Fujian province, Zhangzhou NPP have planned six units arranged as an inverted L-shaped layout along the mountain ridge. The following Figure 2 shows the Zhangzhou NPP plant, with Unit 1 and Unit 2 in the foreground.



Figure 2. An aerial view of Zhangzhou NPP.

4.1 Site Requirements during Construction

The HPR1000 PWR can be deployed at both inland and on-coast sites.

The HPR1000 plant is comprised of the Nuclear Island (NI), the Conventional Island (CI) and the Balance of Plant (BOP). The NI includes: the Reactor Building (NB), the Fuel Building (NF), the Electrical Building (NE), the Safeguard Building (NL, NR), the Nuclear Auxiliary Building (NX), the Access Building (NA), the Connection Building (NN), the SBO Diesel Generator Building (NH), the Fire Protection Pump Station for NI (NP), and the Emergency Compressor House (NC), as shown in the following figure 3.

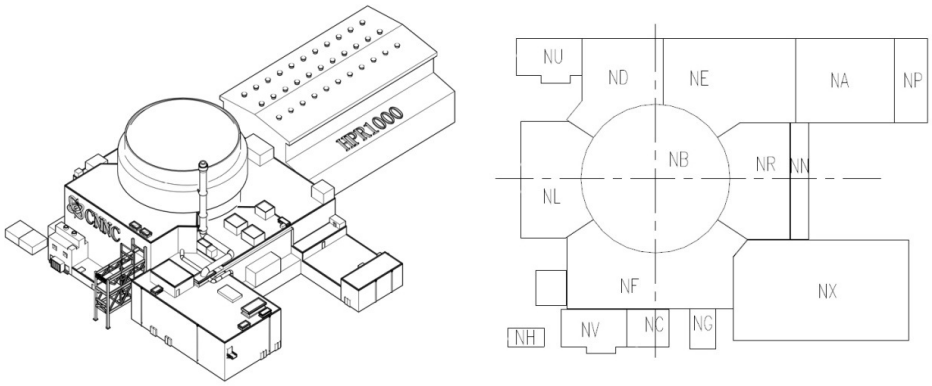


Figure 3. The layout of Nuclear island and conventional island in HPR1000

The NI layout follows the principles of general arrangement:

- The Reactor Building is the core building around which the unit is organized, and the equipment of the safety systems should be located close to the Reactor Building;
- The Safeguard Building and Electrical Building should be located close to the Reactor Building to facilitate the arrangement of pipes and cables; the position of the Fuel Building is related to the fuel transfer tubes.

The Nuclear Island is structurally designed to meet seismic Category I requirements. An integral raft supports the Reactor Building, the Fuel Building, the Electrical Building and the Safeguard Building to

improve seismic capacity.

The arrangement of structures, systems and components important to safety is designed to decrease the possibility and consequences of fires and explosions caused by external and internal events.

A shielding shell is provided to resist the impact of large commercial aircraft crash, while complete physical separation of redundant facilities is also employed.

The plant arrangement provides separation for radioactive and non-radioactive equipment and provides separate pathways to these areas for personnel access. Moreover, the arrangement ensures the operation, inspection, maintenance and component replacement when necessary with high efficiency to minimize radiation exposure.

4.2 Site Considerations during Operation

During normal operation, the HPR1000 nuclear power plant consumes water resources from the surrounding area of the site, mainly including process water and domestic water. Process water includes seawater and freshwater according to the different water quality requirements of specific process. Seawater is mainly used for condenser cooling, auxiliary equipment cooling, essential service water, drum screen flushing water, and the chlorination station. Among these, the first four types of water are directly returned to the sea with unchanged water quality and do not consume seawater, accounting for approximately 99.92% of the total water intake; the actual seawater consumption accounts for only about 0.08% of the total water intake. Freshwater is generally sourced from surface water in reservoirs around the site.

To protect the environment around the site from pollution and prevent workers and residents near the site from receiving excessive radioactive radiation during normal plant operation, the HPR1000 adopts advanced radioactive waste treatment processes. Through the radioactive waste treatment system, radioactive waste is collected, treated, monitored, and temporarily stored. The discharge of treated gaseous and liquid effluents meets applicable standards, ensuring that the impact of radioactive substances released by the nuclear power plant on humans and the environment is maintained at the "As Low As Reasonably Achievable (ALARA)" level. The expected annual generation of waste packages per unit is no more than 50 m^3, achieving the goal of waste minimization.

Based on environmental impact assessments for Units 5 & 6 of the Fuqing Nuclear Power Plant and Units 1 & 2 of the Zhangzhou Nuclear Power Plant, the radioactive source term emissions of HPR1000 units satisfy the discharge requirements specified by national regulations. The radiological impact on the public is well below the constraint value of 0.25 mSv/a stipulated in Chinese national standard and the radiological impact on non-human biota around the site also meets the acceptance criteria.

The HPR1000 has sufficient capability to withstand external events that contribute significantly to risk, such as earthquakes. External events are events originating from outside the nuclear power plant, particularly site-related events such as earthquakes, tornadoes, and floods. Given the wide variety of external events and their close site dependence, the HPR1000 has conducted identification, screening, and enveloping analysis of external events for specific project sites. The results of probabilistic safety analysis for external events indicate that the HPR1000 is adequately robust against external events with significant risk contributions. For seismic events, the standard design of the HPR1000 adopts a peak ground acceleration of 0.3 g for both horizontal and vertical seismic inputs, enveloping the geological conditions of different sites.

The emergency planning zone (EPZ) of the HPR1000 is site-specific. To define emergency planning zone, both design basis accidents and severe accidents are considered, so that emergency preparedness within

the established EPZs can address different accident consequences. Accidents with extremely low probability may be excluded when defining the emergency plan, so as to avoid unreasonable economic burdens resulting from an excessively large EPZ. During the analysis of the EPZ for HPR1000 units, typical design basis accidents and severe accidents are selected for off-site dose consequence assessment. For severe accidents, full accident spectrum weighting is applied, taking into account the occurrence frequency of each release category and its corresponding source term release magnitude. For a specific site, it is recommended that the off-site EPZ for HPR1000 units be set with an inner zone boundary of 5 km, an outer zone boundary of 10 km, and an ingestion emergency planning zone boundary of 50 km.

4.3 Grid Integration

Operating plan of HPR1000 plant is load follow operation.

HPR1000 can bear linear load changes of ±5%Pn/min or ±10%Pn step load changes within a load range of 15～100% Pn and the Xe effect could be compensate by boron concentration adjustment.

The plant can bear continuous linear load changes within a load range of 15～100% Pn, while there should be a certain interval for stable operation between two step load changes.

HPR1000 is designed with capability of load follow for day/night cycle during the top 85% fuel cycle and following the processes within a period of 24h to satisfy the requirement of daily grid variations.

- The plant will operate stably at 100%Pn for at least 12h,
- Linearly decrease the power to 50%Pn within 3h,
- Operate at 50%Pn for at least 6h,
- Linearly increase the power to 100%Pn within 3h.

The two major objectives of boron adjustment are to compensate the slowly burn-up effect and to compensate the Xe poison changes. Associate with R rods, boron adjustment can also join in controlling axis power distribution.

The reactor is designed as the reactor would not trip while the plant is separated with the grid (i.e. full-load rejection) under full load or partial load conditions. In this case, the redundant steam could be released to condenser via turbine bypass.

Under the gray rod control mode, the reactor could compensate the reactivity variations caused by turbine generator ±8% load changes under expected load follow operation conditions.

5. Technical NSSS/Power Conversion System Designn

SUMMARY FOR BOOKLET

The Nuclear Steam Supply System (NSSS) of HPR1000 embody an evolutionary design based on proven three-loop Pressurized Water Reactor (PWR) technology. Key primary circuit components—the Reactor Pressure Vessel (RPV), Steam Generators (SG), and Pressurizer (PZR)—feature increased volumes compared to previous designs. These larger volumes provide increased inventory and response time in the analysed accident scenarios, such as prolonged core coverage during Loss-of-

Coolant Accidents (LOCA), smoother mitigation of operational transients to avoid reactor trips, and extended coping times for accidents like a total loss of feedwater. Dedicated systems, including a depressurization system on the pressurizer, are incorporated for severe accident management.

The reactor core is designed for 3,180 MW thermal power using 177 CF3 fuel assemblies with an 18-month fuel cycle. Reactivity is controlled through three independent means: gadolinium burnable absorbers, Rod Cluster Control Assemblies (RCCAs—both black and grey types), and soluble boron in the coolant. An Emergency Boron Injection System is provided for Anticipated Transients Without Scram (ATWS). The defense-in-depth safety philosophy integrates traditional active safety systems for Design Basis Accidents (DBA) with comprehensive passive systems, such as the Passive Residual Heat Removal System and Passive Containment Cooling System, to prevent and mitigate severe accidents and Design Extension Conditions like Station Blackout (SBO).

The secondary side power conversion system centers on a 1500 rpm turbine-generator set rated at approximately 1212 MWe. The main feedwater system ensures a reliable water supply to the steam generators under all operational modes, utilizing redundant 50% capacity pumps.

The reactor containment of HPR1000 is a double-wall structure. The inner, pre-stressed concrete structure with a steel liner is designed to withstand internal accidents and maintain leak-tightness. The outer reinforced concrete shell protects against external hazards like aircraft crashes. The annular space between them is kept under negative pressure to filter any potential leakage. Both active (Containment Spray) and passive (Passive Containment Heat Removal) systems ensure containment integrity under various accident scenarios.

The Electrical System of HPR1000 is robustly designed with Normal and Emergency Power Supply Systems (NPSS/EPSS). It ensures power to safety-related systems through multiple backups, including off-site sources, Emergency Diesel Generators (EDGs), and supplementary diesel generators for prolonged SBO conditions.

The Instrumentation and Control (I&C) system is fully digital, structured across four traditional levels (process interface to site management). A key design principle is defense against Common Cause Failure (CCF), achieved by employing two diverse technology platforms (SPPA-T2000 and TELEPERM XS) for equipment and software. The human-machine interface is centered on a main control room designed in accordance with human factors engineering principles. It features four identical computerized workstations (three for operators, one for the shift supervisor) and is designed to support operators effectively from normal operations through design extension conditions.

5.1 Primary Circuit

The Reactor Coolant System (RCS) configuration of HPR1000 is that of a conventional PWR three-loop design and can be considered well proven (See Figure 4). The sizing of the HPR1000 reactor pressure vessel (RPV), steam generator (SG) (especially secondary side) and pressurizer (PZR) increases in the respective volumes compared to current 3-loop PWR designs.

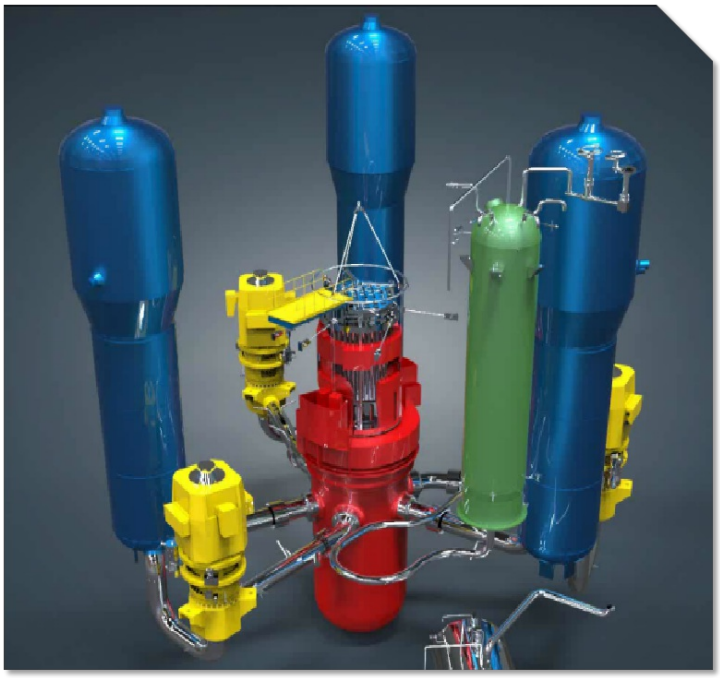


Figure 4. The 3-loop configuration of RCS in HPR1000

In the RPV design, the free water volume between the level of the reactor coolant lines and the top of the active core is increased in order to improve the mitigation of LOCA (smaller breaks) by prolonging the period until the core begins to become uncovered or by minimizing the core uncover depth. The increase of this volume also contributes improving the mitigation of accidents during shutdown conditions, particularly in mid-loop operation (e.g., with loss of Residual Heat Removal System (RHR)), by extending the period for operator action.

The increased pressurizer water and steam volume, with associated pressure and level scaling, reduces the number of load cycles on relevant systems and components. Normal operating transients are mild; the potential for reactor trips is thus minimized.

The large water and steam volumes of the SG secondary side provide the following advantages:

- Smooth pressure and water level changes during normal operating transients, thus further reducing the potential for unplanned reactor trips,
- Significant time delays to mitigate a steam generator tube rupture (SGTR) prior to filling of the SG,
- Extended dry out time beyond the required 30 minutes for the most limiting event of a total loss of feed water supply (including emergency feed water).

Overpressure Protection (OPP) of the Reactor Coolant Pressure Boundary (RCPB) is performed by the PZR safety valves in hot condition and by PZR safety valves and/or RHR safety valves in cold condition with the reactor protection system and associated equipment. The OPP protects the reactor coolant pressure boundary during all anticipated operational occurrences and accidents having the potential for radioactive releases.

The dedicated depressurization system is equipped on the top of PRZ to depressurize the reactor coolant system and prevent the containment failures associated with the high pressure melt ejection (HPME) / direct containment heating (DCH) in the case of a severe accident.

The reactor vessel high point vent system is designed to remove non-condensable gases from the reactor vessel head during accident condition, so as to avoid adverse impact to the heat transfer of reactor core

caused by the non-condensable gases.

5.2 Reactor Core and Fuel

The reactor core of HPR1000 is designed to generate 3,180 MW thermal power with an average linear power density of 181.2 W/cm. The reactor core consists of 177 fuel assemblies made of fuel rods containing uranium dioxide (UO_2) fuel.

The core design provides an operating fuel cycle length of 18 months with low leakage loading patterns and also has flexibility of extending the cycle length. In addition, there are three independent means employed for the core reactivity and power distribution control which are burnable absorber of gadolinium (Gd_2O_3) poisons, rod cluster control assemblies (RCCAs) and soluble boron absorber. The design report states that a thermal margin greater than 15% is provided.

The reactor will load with CF3 fuel assembly. The CF3 fuel assembly is composed of 264 fuel rods arranged within a 17×17 supporting structure. The supporting structure (or skeleton) is composed of 24 guide thimbles, 1 instrumentation thimble, a top and a bottom nozzle, 8 grids and 3 mid-span mixing grids.

The fuel rods containing UO_2 pellets or Gd_2O_3-UO_2 pellets use an advanced zirconium alloy for cladding.

5.3 Fuel Handling

The fuel handling and storage system is designed for handling and storage of fuel assemblies from the receipt of new fuel assemblies to the shipment of spent fuel assemblies.

The major equipment of the system comprises the refueling machine, the fuel transfer system, the spent fuel pit crane, the auxiliary crane, the new fuel elevator, the fuel storage racks and the handling tools.

The refueling machine is located in the reactor building and transfers fuel assemblies between the core and the fuel transfer system. The fuel transfer system conveys fuel assemblies between the reactor building and the fuel building through the fuel transfer tube. The spent fuel pit crane is used to transfer spent fuel assemblies within the fuel building. The new fuel storage racks are used for the storage of new fuel assemblies out of water. The spent fuel storage racks are used for the storage of irradiated fuel assemblies or spent fuel assemblies underwater.

5.4 Reactor Protection

There are three independent means employed for the core reactivity and power distribution control which are burnable absorber of gadolinium (Gd2O3) poisons, rod cluster control assemblies (RCCAs) and soluble boron absorber.

The RCCAs are used for shutdown and control purposes to offset fast reactivity changes. A rod cluster control assembly is comprised of a group of individual neutron absorber rods fastened at the top end to a common spider assembly. There are two types of rods cluster control assemblies:

- Black RCCA containing 24 absorber rods;
- Grey RCCA which contains only 12 absorber rods and 12 grey rods.

The absorber materials used in the control rod design are Ag-In-Cd alloy extruded bars. The absorber materials are essential "black" to thermal neutrons and have sufficient additional resonance absorption to significantly increase their worth. The extruded Ag-In-Cd bars are sealed in cold-worked stainless steel tubes. The absorber stack in the grey rod is replaced by a stack of stainless steel.

The coolant contains soluble boron as a neutron absorber. The boron concentration in the coolant has two functions: one is to control slow reactivity changes necessary for compensating Xenon poisoning or burn-up effects during power operation, and the other is for compensating large reactivity changes associated with large temperature variations during cool down or heat-up phases.

In the event of Anticipated Transient Without Scram (ATWS) condition, the emergency boron injection system (REB) is designed to provide rapid boration to the reactor coolant system to bring reactor core to sub-critical state. Thus, the reactor coolant system pressure peak is limited below the maximum allowable pressure. In the event which needs to provide boric water to the reactor coolant system, if normal boration means is not available, the operator can start REB system manually to inject enough boric water to the reactor coolant system.

Reactor protection system is designed to monitor physical variables, under the two situations below. Reactor protection system initiates the Reactor Trip System (RTS) and Engineered Safety Features Actuation Systems (ESFAS):

- The variables exceeds normal operation limits and control system faults simultaneously;
- During abnormal, event, or accident conditions, to prevent exceeding the specified limits for reactor coolant system and reactor core process parameters.

To minimize the effects of Potential common mode failure, Diverse automatic actuation system (DAS) is designed for diversity protection function (including ATWS) and implemented by different software and hardware equipment from protection system.

Traditional and mature engineered safety features against design basis accidents (DBA) are adopted, to implement functions of core cooling, residual heat removal, containment heat removal, including safety injection system, auxiliary feedwater system, containment spray system, containment isolation system, and etc.

Comprehensive prevention and mitigation measures against various beyond design basis (BDBA) scenario or severe accident (SA) phenomena, including high pressure molten corium ejection, hydrogen detonation, basement melt-through, long-term containment overpressure and station blackout, are incorporated into HPR1000 design. Passive systems are employed as a complement to active safety systems, which can address design extension conditions, thereby enhancing the plant's ability to prevent and mitigate severe accidents. Passive systems includes passive residual heat removal system of secondary side (PRS), Passive Containment heat removal System (PCS) and the passive subsystem of Cavity injection and cooling system (CIS). In design extension conditions such as SBO (Station Blackout), passive systems can achieve core cooling and removal of residual core heat without external power.

5.5 Secondary Side

5.5.1 Turbine generator description

The turbine is a power conversion system designed to change the thermal energy of the steam flowing through the turbine into rotational mechanical work which rotates a generator to provide electrical power.

The turbine is a three-cylinder (one double-flow high pressure cylinder in tandem with two double-flow low pressure cylinders), four-flow exhaust, impulse type, 1500 rpm unit, which is designed for high operating efficiency and maximum reliability. The AC generator is connected directly to the turbine shaft.

The generator is a unit designed for 24kV or 27kV, 3-phase, 50-Hz operation and has a power factor of 0.90. The rated power is 1212MW (Refer to Zhangzhou-1#; depends on site conditions.).

Steam enters the HP turbine through the throttle valves and the governor valves. The governor valves control the flow of steam to the HP turbine cylinder.

The steam passes through the HP turbine impulse stages into the moisture separator and reheater (MSR). The MSR consists of one moisture separator stage and two reheat stages, using two sets of horizontal, cylindrical-shell, combined moisture separator and reheater assemblies. The assemblies are located on each side of the HP cylinder in parallel.

Discharged from the MSR, the steam returns to the low pressure turbine through the reheat stop valves and intercept valves which are mounted on the crossover piping.

Each LP turbine is a double-flow turbine that steam enters into it at the center of the blade path and is admitted into the condenser through the exhaust outlet finally.

5.5.2 Main Feed Water

The main feedwater system is designed to supply the steam generators with adequate feedwater during all modes of plant operation including transient conditions.

The steam exhausted from the turbine is condensed into condensate. Then, condensate is pumped through the gland steam condenser, the condensate polishing system or condensate polishing bypass and four stages low pressure feedwater heaters to the deaerator storage tank by two of three identical 50% capacity motor-driven condensate pumps. Feedwater from the storage tank is pumped by main feedwater pumps to the steam generator finally through two stages high pressure feedwater heaters, each generator's respective flow element, control valve, feedwater isolation valve, and check valve.

There are three 50 percent capacity motor driven feedwater pump sets. During normal operating condition, two feedwater pumps operate in parallel at the various loads dictated by the steam generator and turbine demands. The third feedwater pump is on automatic standby.

Each feedwater pump set includes one booster pump, one motor with coupling and one main feedwater pump.

5.6 Containment/Confinement

The containment of HPR1000 is a double-wall structure located on a common raft (See Figure 5).

The inner containment is a pre-stressed reinforced concrete structure with a leak-tight steel liner, which is designed to withstand the accident condition, e.g. LOCA inside the containment, and retain radioactive fission products.

The outer containment is a reinforced concrete structure which is designed to withstand the effect of external events, e.g. aircraft crash, external explosion, to protect the inner containment and internal SSCs.

The space between the two walls is called 'annular space', where slight negative pressure is maintained to collect possible leakage from inner containment and filtrate it before discharge by specific ventilation system.

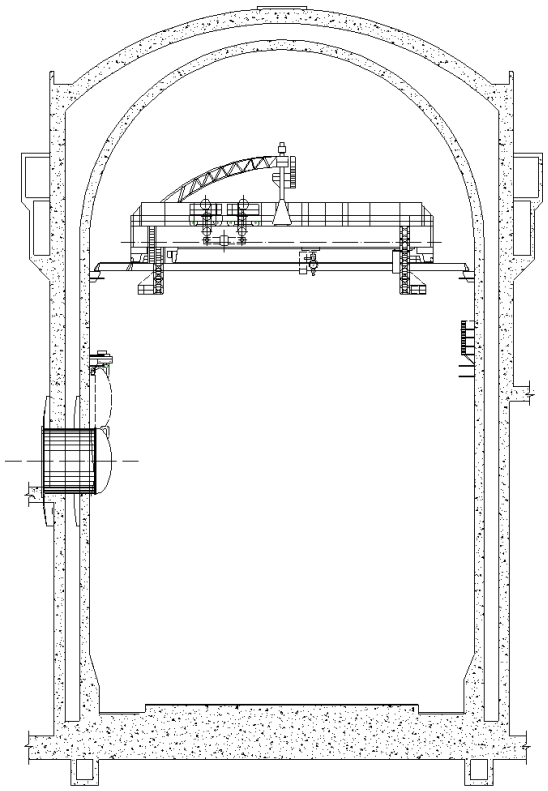


Figure 5. The double wall containment of HPR1000

5.6.1 Inner Containment

The structure of the inner containment is composed of a reinforced concrete base mat, a pre-stressed concrete cylindrical wall and a pre-stressed concrete dome in a half spherical shape. The cylindrical wall, which is 1.3m thick in standard area and thicker in the area around the equipment hatch, has an inner diameter of 46.8 m.

The leak-tightness of the inner containment is ensured by the concrete and the steel liner anchored inside. The leak-tightness requirement for the inner containment is less than 0.3% of the gas contained in the free volume per 24 hours under design basis accident condition. Leak-tightness tests will be performed in pre-operation phase and periodically, to ensure that overall leak-tight requirement and local leak-tight requirement (leakage through equipment hatch, personal airlocks, penetrations, and etc.) can be fulfilled throughout plant life time.

The design temperature and design pressure of the inner containment are 145ºC and 0.52 MPa (abs) respectively. A sufficient margin is provided due to the large free volume (~86000m3) of the inner containment according to the analysis of the effect on the containment by postulated rupture of the primary or secondary loop.

The structural integrity of the inner containment is ensured by the thermal inertia of internal concrete structure, and by the operation of Containment Spray System (CSP), which is actuated when the pressure increases to a certain threshold and quickly reduces the pressure and temperature significantly by spraying cooling water. Nevertheless, under station blackout (SBO) condition when CSP is not available, the Passive Containment heat removal System (PCS) will perform the cooling function and maintain the pressure inside the inner containment within an acceptable value. Thus the integrity of the inner containment can be guaranteed by both active and passive features.

Countermeasures are taken to ensure that the inner containment maintains integrity under severe accident

conditions. High-pressure ejection of molten debris causing containment direct heating can be avoided by fast depressurization of the primary system. The cavity injection of cooling water cools the outer surface of the reactor pressure vessel (RPV) to maintain the molten debris retain inside RPV, so that molten-core concrete interaction (MCCI) will not occur to jeopardize the containment integrity. Accumulation of hydrogen will be controlled by passive autocatalytic recombiners either in design basis conditions or in severe accident conditions.

5.6.2 Outer containment

The outer containment is a reinforced structure with a cylindrical wall, a shallow-spherical dome and a reinforced concrete basemat which extends beyond the containment to support other structures of nuclear island. The outer containment has an inner diameter of 53m, and different thickness of 1.5m (lower part, covered by other surrounding buildings) and 1.8m (upper part, exposed to exterior environment).

The main function of the outer containment is to protect the inner containment and internal structures from extreme external events such as aircraft crash, external explosion and external missiles. In particular, the upper part of the outer containment exposed to exterior environment is designed to withstand the effect by postulated large commercial aircraft crash. Also, the outer containment serves as part of the heat exchange tank structure for passive cooling systems. The three heat exchange tanks locate at the same elevation in exterior surface and form a ring structure supported by the outer containment.

The space between the inner and the outer containment with 1.8 m width is called 'annular space', where a slightly negative pressure is maintained by the operation of the annulus ventilation system, thus to collect and filtrate possible leakage through the inner containment before final discharge to the environment. To ensure the negative pressure of the annular space in all circumstances, certain leak-tightness requirement should be satisfied by the outer containment and the penetrations.

5.7 Electrical, I&C and Human Interface

5.7.1 Electrical systems

Electrical Power Distribution System (EPDS) provides power to four types of auxiliaries: the Plant Unit Auxiliaries, the Permanent Auxiliaries, the Emergency Power-supplied Auxiliaries and BOP Auxiliaries. There are only a few non safety class safeguard important equipment auxiliaries and all safety class auxiliaries belong to Emergency Power-supplied Auxiliaries.

Generally, the EPDS is designed into two types: Normal Power Supply System (NPSS) and Emergency Power Supply System (EPSS). NPSS is divided into six 10kV MV busbars, four for Plant Unit Auxiliaries and two for Permanent Auxiliaries. EPSS is divided into two 10kV MV busbars for Emergency Power-supplied Auxiliaries.

During normal operation of Nuclear Power Plant (NPP), all of the auxiliaries of NPP (including EPSS) are powered by NPSS. During loss of main generator or main off-site power source, the Permanent Auxiliaries and EPSS can be powered by auxiliary off-site power source. During loss of total off-site power (LOOP), EPSS can be powered by on-site power source (the Emergency Diesel Generator sets, EDG). During Station Black Out (SBO), the site supplementary diesel generator set can feed one train of EPSS; The 400V SBO power supply system is composed of two diesel generator sets, one diesel generator is standby for the other. In the event of SBO, the 400V SBO diesel generator set startup automatically and feed hydrotest pump to securing the integrity of the Reactor Coolant System, and also feed some important equipment to insure the available of controls and resident in main control room for 72 hours.

The safety and a few non-safety class safeguard important equipment auxiliaries are powered by EPSS. The safety auxiliaries can protect the reactor safely, limit and reduce the leakage of radioactive materials to environment. Normally, EPSS is supplied by main off-site power source; when loss of main off-site power source, it can switch to auxiliary off-site power source automatically; when LOOP happened, EPSS can powered by EDG for 7 days; when SBO happened, the one of two trains of EPSS can connect to site supplementary diesel generator set manually and be powered for 7 days.

The DC and uninterruptable power supply(UPS) system is part of the EPSS which supplies continuous and reliable low-voltage DC and AC electrical power for I&C auxiliaries and other equipment. There are four UPS groups (one per group) for reactor protection system and two trains for other I&C auxiliaries consisting of batteries, battery chargers, inverters, and associated distribution switchboards.

5.7.2 I&C architecture

The I&C system overall architecture is identified in Figure 6. The following traditional four levels are used for structuring the whole I&C system:

- Level 0: process interface level. It includes the measuring equipment (sensors, transmitters, limit switches) and the actuator interface equipment (pilot valves and their associated interface relay, electro-pneumatic converters, drive actuators, switchgear, power electronic equipment)
- Level 1: control and protection level. It includes signal conditioning and processing equipment in charge of controlling and monitoring the different plant systems
- Level 2: operation and information management level. It includes conventional devices and computerized equipment allowing the plant personnel to operate the plant (manual control and information means), to monitor plant status and to perform service operations on the plant I&C
- Level 3: site management level. It includes computer equipment supporting site management applications and communication with off-site facilities

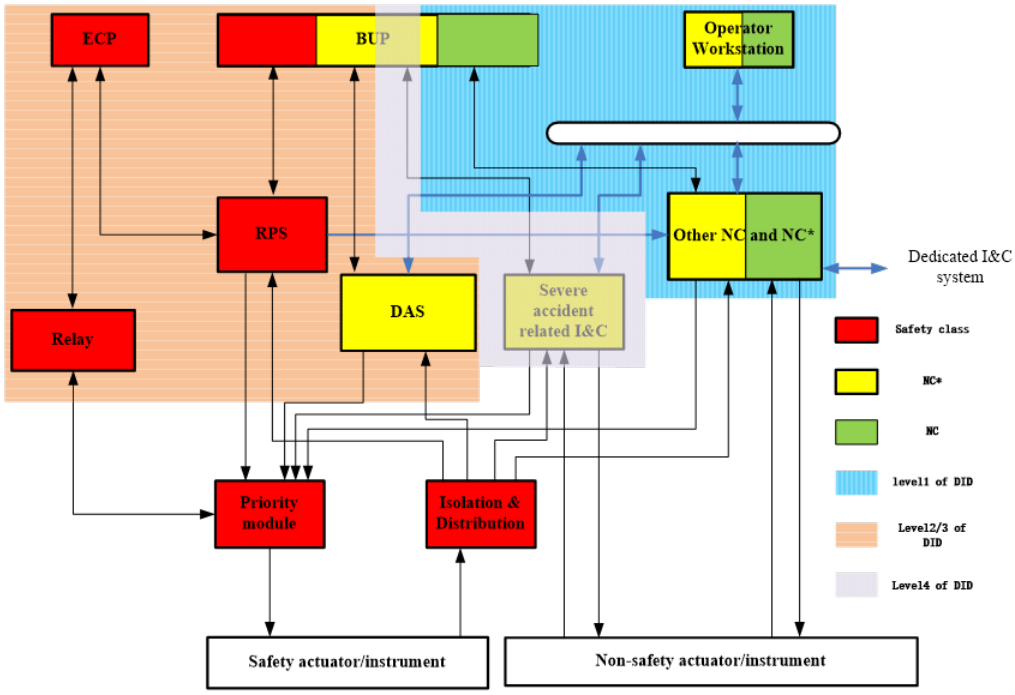


Figure 6. The I&C system overall architecture of HPR1000

HPR1000 I&C system is based on digital technology. The platforms may be different according to different projects. Following is an example of digital platforms application in HPR1000:

- TELEPERM XS, developed and maintained by AREVA,
- SPPA-T2000, developed and maintained by Siemens.

For the purpose of reducing the risk of Common Cause Failure (CCF) impacting different lines of defense, taking benefit of the two diverse technologies SPPA-T2000 and TELEPERM XS. The design features of the system software, libraries of the application software, and main hardware modules for the system platforms TELEPERM XS and SPPA-T2000 are different. Therefore, credit is taken from this equipment and software diversity for the design of the I&C architecture.

5.7.3 Human Interface

The control room is the central point of the I&C system, the most concentrated location for human interfaces, and the center where operators use I&C system equipment installed in the control room to monitor and control the process variables of the NPP. The control room provides operators with the necessary human interfaces, relevant information, and equipment to achieve the operational goals of NPP, and supports operators in mastering operating status, making correct decisions, taking necessary measures in a timely manner, during normal operation, anticipated operational occurrences, design basis accidents, and design extension conditions. The control room system of HPR1000 includes the main control room, remote shutdown station, technical support center, etc.

Firstly, the design of the control room system meets design criteria such as radiation protection performance, fire resistance, seismic performance, human factors engineering principles, and reliability. On this basis, the control room system of HPR1000 has undergone a series of design optimizations and innovations: the design extension conditions are taken into account to functional design, environmental design, and related supporting system design, including operational requirements under severe accident conditions and the habitability requirements of the control room area; it also adopts an advanced operational document system, optimized operational value configuration, and optimized large-screen configuration design to better match operational requirements; human factors engineering principles are integrated into the control room design, and comprehensive verification and validation activities are conducted to ensure that the control room design meets regulatory standards while having a friendly human interface.

The main control room provides operators with all the information and control measures required for operation during normal operation, anticipated operational occurrences, and design basis accidents. The main control room of HPR1000 is equipped with four completely identical computerized workstations, including three operator workstations in the front, which manage, control, and complete all operational tasks according to the requirements of the shift supervisor, and one shift supervisor workstation, serving as the commander during normal operation and the coordinator during accident conditions. The main control room of HPR1000 is located in the electrical building of nuclear island. The main control room of the first HPR1000 unit, Fuqing Unit 5, is shown in the Figure 7.



Figure 7. The main control room of Fuqing Unit 5.

5.8 Unique Technical Design Features (if any)

6. Technology Readiness

SUMMARY FOR BOOKLET

The HPR1000 is an evolutionary pressurized water reactor technology developed based on the experience gained from the design, construction, commissioning and operation of NPPs in China. Its maturity has been assessed through R&D programmes and regulatory reviews.

The design's compliance with stringent international safety standards was further ensured via independent assessments, including an IAEA Generic Reactor Safety Review (2013-2014), reviews by China's National Nuclear Safety Administration (NNSA), and a peer review by the China Nuclear Energy Association.

This mature technology has been successfully deployed. The demonstration project, Fuqing Unit 5, began construction in May 2015 and achieved commercial operation in January 2021, with a reported construction period of 68.7 months. Following this success, the HPR1000 has entered a phase of batch construction and operation within China.

Building on this ready and operational foundation, an upgraded design, the AHPR1000, is now in the design stage.

6.1 Deployed Reactors

HPR1000 is an evolutionary design based on the mature PWR technology adopted by most of the existing NPPs. Most technologies have been proved mature over decades of plant operation. Comprehensive R&D programs have been carried out in developing HPR1000 in early stage. The verification process and results are a part of licensing materials (Safety Analysis Report) which have been reviewed by the regulatory bodies of China

During R&D and design phases of HPR1000, independent technical assessment activities are conducted to ensure that HPR1000 design meet the latest regulatory requirements from both China nuclear safety authority and other international safety authorities, mainly including:

1. IAEA Generic Reactor Safety Review (GRSR) performed for ACP1000 during 2013-2014;
2. Safety review during HPR1000 R&D phase by NNSA (National Nuclear Safety Administration of) China, who issued preliminary evaluation points in Mar. 2012;
3. Peer review of HPR1000 basic design by CNEA (China Nuclear Energy Association) in Apr. 2013;
4. Pre-qualification by NA-SA (Nucleoeléctrica Argentina S.A.) in 2011.

Fujian Fuqing NPP Unit 5&6 is the demonstration project of HPR1000. On May 7, 2015, the Construction Permission of the project was issued by NNSA after several rounds of review activities. As the first HPR100 unit to be deployed, First Concrete for Unit 5 was poured on the same day. On January 30, 2021, the Fuqing Unit 5 was put into commercial operation. With a total construction period of 68.7 months, it set a record for the shortest construction time for the world's first third-generation nuclear power unit (Unit 6 was put into operation on March 25, 2022).

In China, the HPR1000 nuclear power technology has achieved batch construction and operation. The table below shows the deployed HPR1000 PWR projects.

Country	NPP	Number of units
China	Fuqing	2 units in operating
China	Fangchenggang	2 units in operating 2 units approved
China	Zhangzhou	2 units in operating 2 units under construction
China	Taipingling	1 unit in operating 3 units under construction
China	Sanao	1 unit in operating 2 units under construction 1 unit approved

China	Changjiang	2 units under construction
China	Lufeng	2 units under construction
China	Shidaowan	2 units under construction
China	Ningde	2 units under construction
China	Jinqimen	2 units under construction
China	Zhaoyuan	1 unit under construction 1 unit approved
China	Sanmen	2 units approved
China	Taishan	2 units approved
China	Xiapu	2 units approved
China	Xuwei	1 unit under construction 1 unit approved
Pakistan	Karachi	2 units in operating
Pakistan	Chashma	1 unit under construction

6.2 Reactors under Licensing Review

As shown in the table above, 14 HPR1000 units in China have been approved and are currently in the pre-licensing stage.

6.3 Reactors in the Design Stage

On the basis of HPR1000, AHPR1000 is a derivative design under development incorporating design modifications relative to HPR1000, which caters to the future low carbon energy market.

7. Safety Concepts

SUMMARY FOR BOOKLET

The safety design of the HPR1000 is fundamentally structured around the Defense-in-Depth (DiD) principle across five levels to ensure the control of reactivity, heat removal, and confinement of radioactive material. This is implemented through a combination of high-quality, conservative design for prevention; robust systems for controlling operational occurrences; and redundant active safety systems (like Safety Injection and Containment Spray) to manage Design Basis Accidents (DBA). A key objective is the Practical Elimination of Large Radioactive Releases, with a strong focus on maintaining containment integrity under all circumstances.

Building upon the foundation of mature active systems, the design is complemented by a suite of passive safety systems specifically for severe accident prevention and mitigation, significantly enhancing plant resilience. These passive features—including the Passive Residual Heat Removal System (PRS), Passive Containment Cooling System (PCS), and Cavity Injection System (CIS)—are engineered to function during extreme scenarios like a complete Station Blackout (SBO), providing a critical grace period without reliance on AC power.

Furthermore, the design incorporates margins against external hazards (e.g., earthquakes, aircraft crash) and ensures exceptional reliability through diversity and redundancy in safety systems and power supplies. This includes multiple off-site and on-site power sources, culminating in passive systems with dedicated batteries, thereby extending plant autonomy and supporting severe accident management.

7.1 Safety Philosophy and Implementation

The design of HPR1000 is subject to the up-to-date requirements of the Chinese or international nuclear safety codes and guides.

The fundamental safety functions to be ensured in HPR1000 are: 1) control of the reactivity, 2) removal of heat from the core and the spent fuel, and 3) confinement of radioactive materials and control of operational discharges, as well as limitation of accidental releases.

In order to achieve the above safety functions, the concept of defense-in-depth is performed throughout all the safety-related actives to ensure that they are subject to over-lapping provisions.In the design of HPR1000, DiD is well implemented and structured in five levels as required in IAEA documents. Should one level fail, the subsequent level comes into play.

The purpose of the first level of defence is to prevent deviations from normal operation and the failure of items important to safety. This leads to requirements for high quality and reliability of the plant, achieved by

a broad range of conservative provisions taken through the choice of site, design, manufacturing, construction, commissioning, operating and maintenance. The objective of the first level of defence is achieved in HPR1000 with (but not limited to) the following considerations and provisions:

- Investigation and analysis of external factors caused by nature and human activity affecting the plant;
- Proper design management to control the capability of items important to safety, including qualified design personal and quality assurance programme;
- Use of proven technology, with new technology based on appropriate testing and scaling up;
- Safety classification of SSCs, to select and define the appropriate codes and standards for design, construction, manufacture, installation, operation and maintenance;
- Adequate safety margins for the design of safety related items, e.g. 177-fuel-assembly core with lower liner power density to ensure sufficient core thermal-hydraulic margin;
- Qualified material and fabrication for safety related items, e.g. the improvement of the material and fabrication for RPV contributing the extended lifetime of 60 years;
- Conservative seismic structural design with seismic input of 0.3 g peak ground acceleration;
- Protection against the crash of large commercial aircraft for safety related buildings;
- Improved operational stability with enlarged pressurizer volume, which provide better tolerance for power transient;
- Minimization of manual operator actions and use of adequate operating procedures to reduce the possibility of incorrect operator actions;
- Definition of operating range of plant variables, and safety limits set up based on conservative analysis;
- Suitable radiation monitoring facilities and radiation protection measures to control the occupational exposure dose below 1 manžSv/reactoržyear.

The purpose of the second level of defence is to detect and control deviations from normal operation in order to prevent anticipated operational occurrences from escalating to accident. The following provisions are incorporated in HPR1000 for the second level of defence:

- Inherent plant features, such as core stability, thermal inertia and negative reactivity coefficient feedback;
- Protection against power transient accidents with the monitoring of rod position, limited speed of rod withdrawal and prevention of inadvertent boron dilution;
- Reliable digital instrumentation and control system providing necessary control, protection and monitoring information for plant operation;
- Diverse actuation system providing automatic and manual control function, to reduce the risk caused by common failure of digital protection system;
- Diagnostic tools and equipment such as automatic control systems provided to actuate corrective actions before reactor protection limits are reached;
- Advanced in-core instrumentation for monitoring of the neutron flux and temperature of reactor core;
- Advanced Main Control Room, with selected parameters monitored and clear indication of plant status, taking advantage of the man-machine interface and human factors engineering;
- Regular in-service inspection and periodic testing of systems and plant components, with easy access of equipment for inspection;
- Modern non-destructive testing (NDT) methods to recognize possible defects;
- Means for LBB concept implementation;
- Feasible repairs and replacement for all the major equipment.

For the third level of defence, it is assumed that the escalation of certain anticipated operational

occurrences or postulated initiating events might not be controlled at a preceding level and that an accident could develop. This leads to the requirement that inherent and/or engineered safety features, safety systems and procedures be provided to control the accident within the design basis and prevent evolution towards severe accidents and also to confine radioactive materials within the containment system. The measures taken in HPR1000 at the third level of defence are itemized as below:

- Safety injection system to provide emergency core cooling, with active middle-head and low-head injection and passive accumulator injection, and system configuration improved based on PSA results;
- Containment spray system to reduce the temperature and pressure in the containment after accident, with optimization in the capacity, layout and equipment;
- Provision of auxiliary feedwater system to supply feedwater to the secondary side of steam generator with diverse power sources, i.e. motor-driven pump and turbine-driven pump;
- In-containment refueling water storage tank, with better protection for external hazard and saving the need for water source switch;
- Reliability of the engineered safety systems are ensured with principles of redundancy, diversity and independence;
- Redundant, diverse and reliable shutdown systems, with one of them capable with margin and single failure criterion;
- Extension of nonintervention period to 30 minutes.

The purpose of the fourth level of defence is to mitigate the consequences of accidents that result from failure of the third level of DiD. The most important objective for this level is to ensure the confinement function, thus ensuring that radioactive releases are kept as low as reasonably achievable. Various means are established in HPR1000 aiming at preventing severe accidents and mitigating their consequences:

- Active and passive cavity injection system, to achieve IVR of melt core by cooling the external surface of RPV and maintain the integrity of its lower head;
- Passive residual heat removal system of secondary side, to remove the core residual heat during SBO accident;
- Passive containment heat removal system, used for long-term heat removal of containment in design extension conditions so as to reduce the pressure and temperature of containment to maintain its integrity;
- Containment hydrogen recombination system, to reduce the hydrogen concentration of containment atmosphere and prevent hydrogen detonation by passive hydrogen recombiner;
- Rapid depressurization system of RCS, to suppress the occurrence of high pressure core melt;
- High-point venting system of RPV, to discharge the non-condensable gases in the upper head of RPV;
- Containment filtration and venting system, to depressurize the containment to ensure the pressure not exceed the bearing limit and filter the radioactive material in the discharge gas;
- Double-shell containment and enlarged containment free volume, to enhance the safety of the last barrier;
- Improved inhabitability of main control room in the condition of severe accident;
- Emergency power supply to important equipment necessary for accident response, and the load of monitoring, control, ventilation, lighting if all the AC powers are lost;
- Emergency feedwater makeup from on-site/off-site water sources to important systems to remove the residual heat of core and spent fuel;
- Severe Accident Management Guideline (SAMG), fully utilizing existing plant capabilities, including available non-safety-related equipment to realize the purpose of accident management.

The purpose of the fifth and final level of defence is to mitigate the radiological consequences of radioactive releases that could potentially result from accident conditions. This requires the provision of emergency response. The provisions implemented in HPR1000 for the fifth level of defence include:

- Well-equipped emergency control center, with reliable communication means and power supply, and improved inhabitability and availability in severe accident conditions;
- On-site emergency plan and procedure;
- Off-site emergency plan consistent with on-site plan;
- On-site and environmental radiological monitoring, including mobile monitoring equipment, to obtain adequate information for consequence estimation;
- Sufficient resources for emergency response, including food and water supply, medical care, engineering and technical support and etc.;
- Emergency preparedness for multiple-units accident and long-term SBO;
- Periodic exercises and drills for emergency response.

A relevant aspect of the implementation of DiD for a nuclear power plant is the provision in the design of a series of physical barriers, as well as a combination of active, passive and inherent safety features that contribute to the effectiveness of the physical barriers in confining radioactive material at specified locations. For HPR1000 at power operation, the barriers typically include the fuel pellet, the fuel cladding, the boundary of the reactor coolant system and the reactor containment.

Structure, systems and components important to safety are capable of withstanding identified initiating events with sufficient robustness, which are ensured by the design criteria such as redundancy, diversity and independence. Active and passive safety design is a significant instance to fulfill the diversity criteria. The rigorous physical separation for redundant safety systems is also achieved.

The target of practically eliminating the large radioactivity release has drawn extensive discussion after Fukushima accident. With the purpose of controlling LRF below 10-7 /reactor·year., the integrity of containment is emphasized in the design of HPR1000, by adopting comprehensive severe accident prevention and mitigation measures against the postulated severe accident sequences identified by probabilistic methods and deterministic methods.

With the target to further eliminate the residual risk, sufficient margins are considered in the design to protect the plant from beyond-design-basis external events such as earthquake, flooding and large commercial aircraft crash. The emergency response capability is also improved by providing mobile pump and mobile diesel generator. The passive systems are able to operate for 72 hours with sufficient volume of storage water and dedicated battery, which is of significance for the extension of plant autonomy period.

7.2 Transient/Accident Behaviour

Traditional and mature engineered safety features against design basis accidents (DBA) are adopted, to implement functions of core cooling, residual heat removal, containment heat removal, including safety injection system, auxiliary feedwater system, containment spray system, containment isolation system, and etc., mainly as shown in Figure 8.

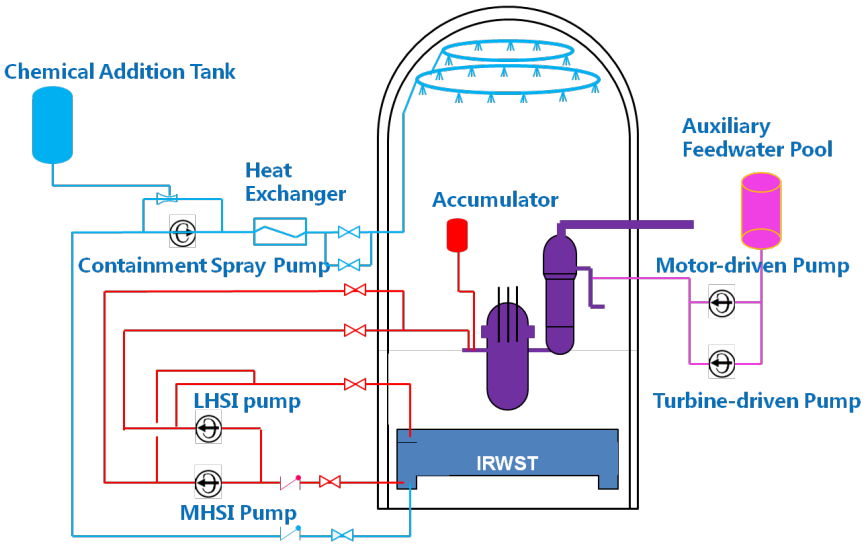


Figure 8. Engineered safety features against DBA in HPR1000

The primary function of the **Safety Injection System (SIS)** following an accident is to remove the decay heat from the reactor core and to provide core cooling by injecting water to the reactor. The RSI consists of three accumulators, IRWST, two MHSI pumps, two LHSI pumps and associated valves, pipes, and instruments. The MHSI and LHSI pumps take suction from the IRWST after LOCA, and inject the boron water to the RCS to improve core cooling and limit core damage and the relief of fission product, and to provide additional shutdown capability.

The **auxiliary feedwater system (TFA)** ensures a sufficient water supply to the steam generators for removal of decay heat from the core in case of loss of coolant accident, or a main steam/feedwater line break, or a total failure of the normal feedwater systems supplying the steam generators. The auxiliary feedwater system is provided with 2×50% motor-driven pumps which are backed up by diesel generator and 2×50% turbine-driven pumps which are supplied with steam produced by the three steam generators. Each train consists of one motor-driven pump and one turbine-driven pump, and close attention is paid to the physical separation of the two trains.

The containment spray system (CSP) comes into operation after an accident that causes an increase of containment pressure and temperature (loss-of-coolant accident or secondary line break accident). The CSP system maintains containment integrity by limiting containment pressure and temperature within the design limits in case of design basis accidents.

The containment spray system is designed to operate over an extended period of time, possibly lasting several months, as long as reactor containment integrity is not ensured. To improve the long-term reliability of the “residual heat removal” function, pipelines are provided to allow the installation of a connection about two weeks after the accident, which enables the containment spray system pumping set to be backed up by the pumping set of the low-head safety injection system.

The **Containment Spray System (CSP)** consists of two identical and physically separated trains, and each train is capable of fully ensuring the spray function. The in-containment refueling water storage tank (IRWST) and the spray additive subsystem are common to both trains. Each train includes a motor-driven spray pump, a heat exchanger cooled by the component cooling system (WCC) water, two spray rings underneath the containment dome and piping. Water is sprayed from IRWST into the containment. The chemical additive is injected into this water to reduce the quantity of airborne fission products (especially iodine) in the containment.

Comprehensive prevention and mitigation measures against various beyond design basis (BDBA) scenario or severe accident (SA) phenomena, including high pressure molten corium ejection, hydrogen detonation, basement melt-through, long-term containment overpressure and station blackout, are incorporated into design as shown in Figure 9.

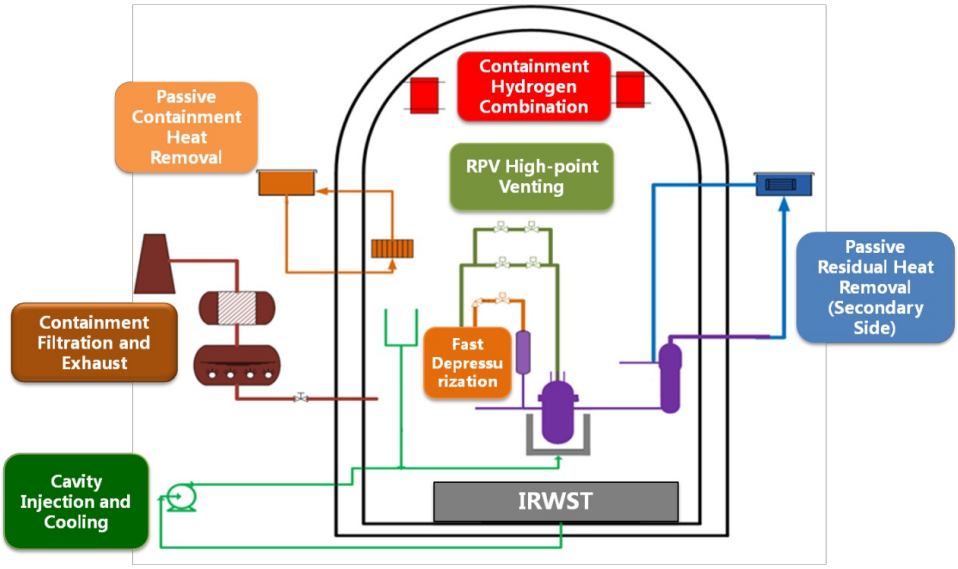


Figure 9. Comprehensive prevention and mitigation measures against BDBA

Passive residual heat removal system of secondary side (PRS) is employed. During the Station Black-Out (SBO) condition overlapped with failure of the turbine-driven auxiliary feedwater pumps, the PRS system passively supplies feedwater to the secondary side, removes the reactor decay heat and the sensible heat of the primary equipment to environment, and maintains the reactor under a safety state in 72 hours without water makeup. The PRS system consists of three trains associated to three SGs for the RCS loops. Each train includes one emergency heat exchanger, two emergency make-up tanks, one accident cooling tank and the necessary piping, valves and instruments.

Passive Containment heat removal System (PCS) is designed to be a passive cooling system which will ensure that the pressure and temperature of containment will not exceed the design limits during the Beyond Design Basis Accident (BDBA) scenarios. The heat will be continuously taken external to containment by water flowed in tubes of heat exchangers to heat exchanger tanks installed outside containment which finally discharge the heat of containment to environment. The water flow is driven by density difference due to temperature difference. PCS consists of three independent trains. Each train is composed of heat exchange tank, heat release tank and two heat exchange loops. Additionally, Heat Exchange Tank is provided as the ultimate heat sink of PRS and PCS. The tank inventory fulfils the requirement of 72 hours passive cooling during post-accident.

During severe accident, the Cavity Injection and cooling System (CIS) can cool the molten corium inside reactor pressure vessel (RPV) by external vessel cooling, to reduce the temperature of the external surface of RPV, and maintain its integrity.

The Cavity injection and cooling system (CIS) consists of an active subsystem and a passive sub-system. The active subsystem of CIS includes two parallel connected injection lines, each equipped with an injection pump. The inlet of each pump is connected to the Safety Injection System (RSI) pump suction line so that the water in IRWST can be injected to the cavity during severe accident; meanwhile, the water of Fire Water Distribution (FWD) is on standby and can be connected to the CIS system. The passive

subsystem is located in the containment. During severe accident and the active subsystem fails, the isolation valves open, then the water in the CIS tank injects to the channel formed between the insulation layer and the external surface of RPV, submerging the RPV bottom gradually for passive cooling.

The function of **dedicated depressurization system** is to depressurize the reactor coolant system and prevent the containment failures associated with the high pressure melt ejection (HPME) / direct containment heating (DCH) in the case of a severe accident. The dedicated depressurization system consists of two parallel discharge line connected to a common specific nozzle on the dome of the pressurizer to meet the redundant requirement. Each line is equipped with two motor-operated valves, a gate valve and a globe valve in series.

The **reactor vessel high point vent system** consists of normal venting sub-system and accident venting sub-system. The normal venting sub-system connects to the top of the reactor vessel head and is equipped a reducer with a flow restrictor and a normal closed globe valve. The accident venting sub-system line connects to the RPV normal venting pipe, through a flow restrictor, and then is separated into two lines, train A and B. Two solenoid globe valves are equipped on both train A and B. The combination of two redundant lines connects to the pressurizer discharge line. The normal venting sub-system is used only in shutdown condition for normal venting of RCS, while the accident venting sub-system is designed to remove non-condensable gases from the reactor vessel head during accident condition, so as to avoid adverse impact to the heat transfer of reactor core caused by the non-condensable gases.

The **Containment Hydrogen Combination System (CHC**), in which 33 passive autocatalytic recombiners (PARs) to be installed, is intended for decreasing hydrogen concentration within the containment atmosphere up to safe limits which may cause hydrogen inflammation during DBA or hydrogen detonation and disturbance of the containment integrity during severe accident. When the hydrogen concentration is up to the threshold for starting of hydrogen removal, PARs will self-motion to decreasing hydrogen concentration within the containment atmosphere up to safe limits.

The **Containment Filtration and Exhaust System (CFE)** can be an option for preventing the pressure of the containment from exceeding its bearing capability thus to ensure its integrity. Besides, the filtration equipment equipped in the depressurization line can filter and retain the radioactive materials vented.

In the event of Anticipated Transient Without Scram (ATWS) condition, the **emergency boron injection system (REB)** is designed to provide rapid boration to the reactor coolant system to bring reactor core to sub-critical state. Thus, the reactor coolant system pressure peak is limited below the maximum allowable pressure. In the event which needs to provide boric water to the reactor coolant system, if normal boration means is not available, the operator can start REB system manually to inject enough boric water to the reactor coolant system.

During accidents, reliable power is necessary for operation of active safety systems and seal of reactor coolant pump, therefore the **diversity of power** is also important for HPR1000 safety operation. HPR1000 has two independent off-site electrical power supply as main power and auxiliary power, if the main off-site power fails, the auxiliary off-site power automatically into service. In case of failure of off-site power, HPR1000 has two diesel generators and one additional diesel generator as on-site emergency power. If the station blackout (SBO) with loss of off-site & on-site power, two SBO diesel generators start into operation and bring the plant into safety shutdown condition. However, even all power above lose, HPR1000 has temporary power to supply power by mobile diesel generators. Besides, HPR1000 has different voltages batteries to supply DC power. Base on the demand of uninterruptible power supply (UPS), two independent 72h DC power systems are to ensure adequate power supply for valves, instrumentation and control of

passive safety systems.

8. Fuel and Fuel Cycle

SUMMARY FOR BOOKLET

The HPR1000 reactor core comprises 177 fuel assemblies (including the CF3 type with 264 fuel rods in a 17×17 structure), adopting an 18-month fuel cycle (68 assemblies reloaded per refueling) with a 24-month long-cycle option. After reaching specified burn-up, spent fuel cools in plant pools for years before transfer to reprocessing plants—where most of valuable uranium and plutonium (excluding oxygen) are recovered and recycled, while high-activity long-lived wastes undergo glass matrix conditioning for dedicated storage. Supported by advanced fuel management, the design optimizes resource use, reducing natural uranium consumption and refining waste, leveraging CNNC's full fuel cycle expertise.

8.1 Fuel Cycle Options

The reactor core consists of 177 fuel assemblies made of fuel rods containing uranium dioxide (UO2) fuel.

The HPR1000 reactor load with CF3 fuel assembly. The CF3 fuel assembly is composed of 264 fuel rods arranged within a 17×17 supporting structure. The supporting structure (or skeleton) is composed of 24 guide thimbles, 1 instrumentation thimble, a top and a bottom nozzle, 8 grids and 3 mid-span mixing grids.

18 months fuel cycle length is adopted in HPR1000 design, while for each refueling, 68 of the 177 fuel assemblies in core will be reloaded. Longer fuel cycle strategy capability such as 24 months is reserved.

The application of long cycle refueling benefits the overall availability of the plant and reduces the power generation costs.

After reaching a certain burn-up, the fuel is sent to the spent fuel pools of the nuclear power plants for a few years of cool down. Then the spent fuel elements are transferred to the spent fuel pool of the reprocessing plant. In the reprocessing plant, materials of the spent fuel which can still be valued, such as plutonium and uranium, which represent about 95% of the total mass (excluding oxygen) are separated and recycled as fuel. Materials which cannot be valued, such as high activity-long life wastes, are conditioned in a glass matrix and stored in a specific facility.

The plutonium originating from reprocessed spent fuel is recycled in the form of mixed uranium and plutonium oxides fuel (MOX). Presently, MOX spent fuel is not further reprocessed, although representing a potential economic value. The highly radioactive accumulated fission products prevent immediate reprocessing.

The uranium originating from the reprocessed spent fuel can either be re-used in PWRs after being re-enriched in U235, or stored for further utilization in fast breeder reactors.

In some countries, the spent fuel may be stored in an interim storage facility, pending a decision about either

reprocessing, if it is allowed by the law of such countries at that time, or the setting up of a final geological waste disposal facility.

CNNC has proven experience in uranium exploration, mining, conversion, enrichment, advanced fuel fabrication and trading, spent fuel dry or wet storage, spent fuel reprocessing, etc.

8.2 Resource Use Optimization

HPR1000 has employed the advanced fuel management scheme and long fuel cycle, which lowers the neutron leakage and improves the neutron economy.

The increased discharge burnup of spent fuel maximizes the output energy per initial uranium, and natural uranium consumption is minimized, as well as the waste from the refining process.

8.3 Unique Fuel/Fuel Cycle Design Features

9. Safeguard & Physical Security

SUMMARY FOR BOOKLET

This chapter outlines the safeguards and physical security measures of the HPR1000 nuclear power plant. In terms of safeguards, the plant uses low-enriched uranium (≤5wt% 235U) as its principal fissile material, with strict administrative control over fuel assemblies and a design unsuitable for weapon-grade plutonium production. For physical security, the Physical Protection System (PPS), compliant with Chinese codes/standards and IAEA recommendations, adopts defense-in-depth and balanced protection principles, monitors all kinds of illegal intrusions and make a response timely, quickly and accurately to intrusion actions from adversary or adversarial group by dividing the site to several areas with physical barriers, setting intrusion alarm equipment, video cameras and communication equipment, and by controlling the access of personnel and vehicle to different areas.

9.1 Safeguards

The HPR1000 plant does not present any unique issues relative to other light water reactors with regards to proliferation and commercial nuclear power plants.

The principal fissile material used in HPR1000 reactor is low enriched uranium (LEU), with the 235U enrichment of no more than 5wt%, which is nearly useless for a nuclear weapon.

A fuel assembly is under strictly administrative control from its birth to final reprocessing or disposal. And technically, a fuel assembly is much too heavy to be handled and transported without specific equipment. With the administrative procedures and its intrinsic characters, the fresh fuel is nearly no risk of diversion or misuse.

Plutonium is transformed from the uranium in the reactor under irradiation. In order to obtain the weapon grade plutonium, the burnup of the fuel must be extremely low because the quality of the plutonium will decrease with respect to the burnup, which means the reactor must operate with high frequency of refueling, which can be easily detected.

And the spent fuel must be processed in a specific reprocessing plant to extract the plutonium. Reprocessing is an extreme complicated process, and necessitates a highly sophisticated infrastructure, which cannot be done without notice.

HPR1000 is designed as a commercial light water reactor for electricity production, and is suitable for operation with long fuel cycle such as 18 months, which makes it nearly impossible for weapon grade plutonium producing using HPR1000 reactors.

9.2 Security

The Physical Protection System (PPS) of HPR1000 is designed according to the characteristics of HPR1000 and based on the China Codes and standards, also referring to the IAEA recommendations.

PPS has the function of detection, delay and response, and follows the basic principles of defence in depth and balanced protection. By dividing the site to several areas with physical barriers, setting intrusion alarm equipment, video cameras and communication equipment, and by controlling the access of personnel and vehicle to different areas, PPS can monitor all kinds of illegal intrusions and make a response timely, quickly and accurately to intrusion actions from adversary or adversarial group. It can ensure a safety and reliable environment for the operation of nuclear power plant (NPP).

The protected targets of PPS mainly include nuclear materials (fresh fuel and spent fuel), important systems and vital equipment, whose destruction would lead to severe nuclear leakage accidents.

Technical protection measures of PPS consist of integrated management system, access control system, intrusion alarm system, video monitoring system, security communication system, patrol system, security lighting system and security power supply system.

The sub-systems of PPS are managed by security integrated management system. So that the sub-systems can be operated reliably and independently, meanwhile the data exchange and linkage control can be implemented quickly and effectively as well.

The access control system is used for authorization and access management of personnel, vehicle and goods which enter or exit each security area.

The intrusion alarm system is used to detect all kinds of illegal intrusion actions.

The video monitoring system is used to monitor the accesses, perimeters and important positions of NPP.

The security communication system is used for communicating among the personnel access, vehicle access, security rooms and Central Alarm Station (CAS).

The patrol system supplies effective management means for guard patrol.

The security lighting system supplies the reliable lighting for perimeters and accesses of each security areas.

The security power supply system provides safe and reliable power supply for all equipment of PPS.

9.3 Unique Safeguards and/or Security Features

None.

10. Project Delivery & Economics

SUMMARY FOR BOOKLET

The HPR1000 project features mature and efficient delivery and operational economics: 70% of detailed design is completed before first concrete, with most key equipment localized in China for a reliable supply chain; advanced construction technologies enable construction schedule of 62~69 months (from FCD to provisional acceptance), exemplified by the 68.7-month record of the Fuqing Unit 5 demonstration project. For operation and maintenance, it adopts load follow operation to adapt to grid variations, with a designed availability factor of no less than 90%, supporting stable long-term operation.

10.1 Project Preparation and Negotiation

About 70% detailed design would be completed before FCD (First Concrete Date).

Most of main equipment are proven and can be manufactured in China, which provides a economic and convenient equipment supply chain. The mature and strong equipment supply chain of China can support project construction and plant long-term operation.

10.2 Construction and Commissioning

Construction and installation technologies have been developed and applied in HPR1000 project, e.g. modular construction, open-top installation, automatic welding, digital and smart construction.

The construction schedule of a HPR1000 unit depends on the specific project situation, such as the site condition, the industrial organization, equipment supplying localization and local participation conditions. A typical construction schedule of HPR1000 unit is reported approximately 62~69 months, depending on project-specific conditions.

The demonstration project of HPR1000, the Fuqing Unit 5 has put into commercial operation with a total construction period of 68.7 months. And the construction duration of Zhangzhou Unit 1 is 62.5 months.

10.3 Operation and Maintenance

Operating plan of HPR1000 project is load follow operation.

HPR1000 is designed with capability of load follow for day/night cycle during the top 85% fuel cycle and following the processes within a period of 24h to satisfy the requirement of daily grid variations.

- The plant would stable operation at 100%Pn for at least 12h,
- Linear decrease the power to 50%Pn within 3h,
- Operation at 50%Pn for at least 6h,
- Linear increase the power to 100%Pn within 3h.

The reactor is designed as the reactor would not trip while the plant is separated with the grid (i.e. full-load

rejection) under full load or partial load conditions. In this case, the redundant steam could be released to condenser via turbine bypass.

Under the gray rod control mode, the reactor could compensate the reactivity variations caused by turbine generator ±8% load changes under expected load follow operation conditions.

According to the presumed duration of HPR1000 nuclear power plant refueling outage and ten-year outage, and preventive and mitigating measures of unplanned scrams and duration , the goal of HPR1000 units availability factor being no less than 90% could be attained.