Status report – Chinese Supercritical Water-Cooled Reactor (CSR1000)

Overview

Full name	Chinese Supercritical Water-Cooled Reactor
Acronym	CSR1000
Reactor type	Pressurized Water Reactor (PWR)
Coolant	Supercritical Light Water
Moderator	Light Water
Neutron Spectrum	Thermal
Thermal capacity	2,300 MW_{th}
Electrical capacity	1,000 MW_e
Design status	Conceptual Design
Designers	Nuclear Power Institute of China (NPIC)
Last update	12-2015

Description

1. INTRODUCTION

The Supercritical Water-Cooled Reactor (SCWR) nuclear energy system is a combination of the advanced nuclear reactor technology and the updated supercritical fossil boiler technology whose plant efficiency can reach up to 50% (Europe Future I, 2005) recently. Compared with the current water-cooled reactor NPPs whose average plant efficiency is only around 33%, the SCWR has significant economic and technical advantages. The Generation IV International Forum selected SCWR as one of the six most promising Generation IV nuclear energy systems in 2002 after technical investigation and comparison among more than 100 initiative reactor types for several years.

The SCWR operates above the thermodynamic critical point of water (374℃, 22.1MPa), which presents both technical advantages and challenges.

The first advantage of SCWR, as mentioned above, is the high plant thermal efficiency thanks to use of supercritical turbine-generator system. The second one is extraordinary simplification of the primary system, which eliminates the steam generators, pressurizer and main circulation pumps of the PWRs, the inner circulation pumps, steam separators and dryers of BWRs. As for the safety-related advantages, due to the characteristics of the supercritical water, no phase change would occur in the SCWR core under nominal conditions, therefore there is no risk of Departure from Nucleate Boiling (DNB) like in traditional PWRs. The high temperature operation conditions also exclude the use of Zr as fuel cladding material and thus assure no H_2 gas produced from Zr-water reactions under severe accidents, so there is reduced risk of H_2 gas explosion, which has become a worldwide concern for the Water Cooled Reactors after the Fukushima accident.

The most important technical challenge for SCWR is the fuel cladding material and reactor structure material because the materials used in the existing nuclear reactors cannot meet the high temperature requirements for the SCWR operation conditions.

In 2009, the Chinese government entrusted a national SCWR Technical R&D Program (Phase I, 2010-2012) to the Nuclear Power Institute of China (NPIC), the only comprehensive nuclear power R&D, experiment and design base in China. This program was successfully completed in 2012 as required and a number of initiative and original research achievements were made in the aspects of conceptual design, thermal-hydraulic tests and material R&D, with more than 20 patents applied. The first Chinese SCWR conceptual design CSR1000 was established with data mostly obtained in China. Successful completion of this program has made China one of the main players of SCWR R&D in the world like Japan, Europe and Canada. Application of the successive Program Phase II has just finished the technical approval process with positive conclusion.

The fundamental philosophy of the Chinese national SCWR R&D program can be summarized in the three principles as follows:

i) Be in line with the GIF technical objectives,

ii) Focus on technical feasibility study; avoid any “paper reactor”,

iii) Perform the design under support of the reliable test data.

The CSR1000 main technical parameters, core design, key structure design and systems design, etc. were carefully studied along with the material R&D and thermal hydraulic experimental R&D. The design and tests were closely integrated and the internal results feedback optimized the design.

Two test facilities were designed and constructed in NPIC for SCW T/H experiments; one is SCW mechanism loop and the other SCW T/H loop. With the two facilities, the flow and heat transfer experiments have been completed in various flow geometries, circular pipe, circle-annuli, square-annuli and small 2×2 rod bundles. A SCW test database has been compiled covering large range of operating conditions, which serves to develop experimental correlations and assess computer codes for the SCWR design. The SCW corrosion and SSRT loop was also constructed, covering the CSR1000 parameter range (max temperature and pressure 650℃/25MPa, Water flow 2~5 L/h, pH under control etc.). The corrosion and stress corrosion cracking in SCW, mechanical behaviors and ion irradiation on the SCWR candidate materials had been investigated.

The CSR1000 conceptual design with the related test data established by NPIC provides a solid and reliable base for further R&D towards realistic application of the SCWR. The main technical data of CSR1000 are shown in the Appendix.

2. DESCRIPTION OF THE NUCLEAR SYSTEMS

2.1. Main characteristics of the primary circuit

The main characteristics of the CSR1000 primary circuit are as follows:

1) The primary circuit of CSR1000 is a direct-cycle system consisting of a two-pass, thermal neutron reactor cooled and moderated by light water, two primary loops connected with supercritical turbine and feedwater pumps etc. The primary circuit is also interfaced with passive safety features.

2) The core thermal power is 2300MWt with system thermal efficiency 43.5%, leading to system output electrical power around 1000MWe.

3) The primary circuit operates at 25.0 MPa. The feedwater temperature is 280℃, and the average core outlet coolant temperature is about 500℃.

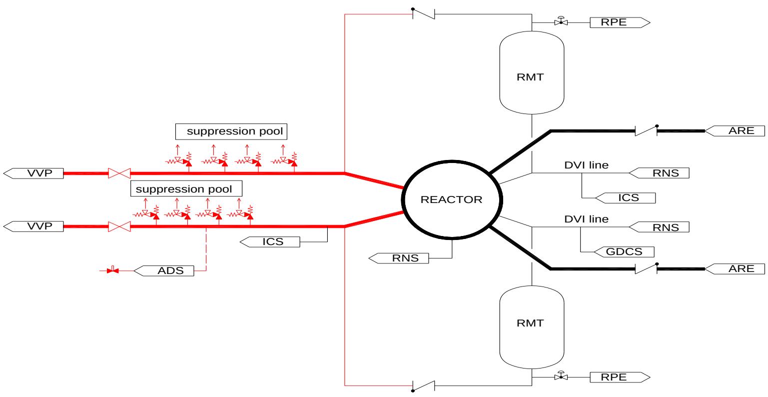


Fig.1 flow diagram of CSR1000 primary circuit

2.2. Reactor core and fuel design

The CSR1000 reactor core consists of 157 fuel assemblies. The core coolant flow rate, 1189 kg/s, is significantly lower than those of current LWRs since the enthalpy rise in the core is much higher than that of traditional LWRs.

The CSR1000 fuel loading and reloading patterns (1/4 symmetric core) are shown in Fig 2. As is shown in Fig.3, a two-pass core arrangement is presented. Fifty-seven FAs are located in the center, signed as “Ⅰ” style fuel assembly; the rest lie in periphery of the core, marked as “Ⅱ”style fuel assembly. In order to meet refueling requirement, the structure of “Ⅰ” style fuel assemblies must be the same as that of “Ⅱ” style fuel assemblies. Considering two-pass core arrangement, moderator water passage and coolant passage must be separated from each other. In addition, moderator water and coolant should have enough flow area, and there is no interference with structure among the different components of fuel assembly.

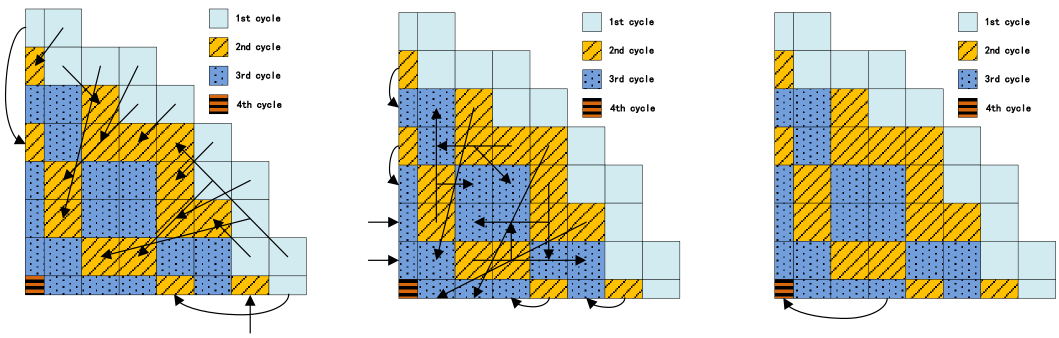
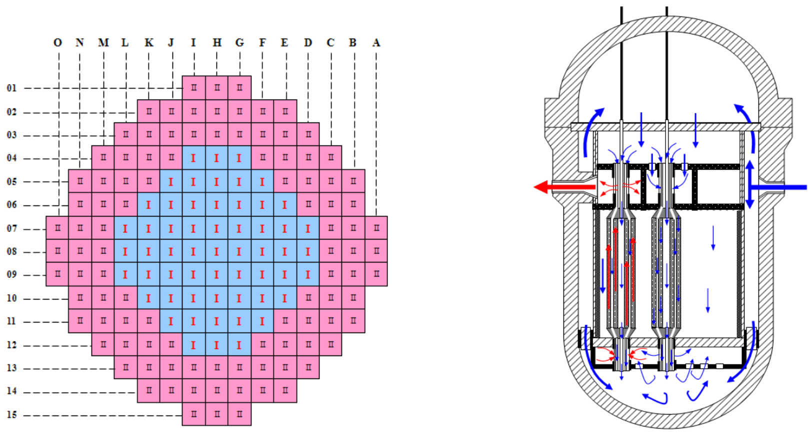


Fig.2 Fuel loading and reloading patterns (1/4 symmetric core)



3(a) 3(b)

Fig.3 Two-pass core design

The U^{235} enrichment for the equilibrium core is 5.6% to achieve the discharge burnup of about 33,000 MWd/tU. After optimization study, the discharge burnup can reach about 45,000 MWd/tU while the U^{235} enrichment for the equilibrium core is about 6.2%. The enrichments are both slightly higher than those of the current LWRs because of higher neutron absorptions of the stainless steel structural materials used for fuel claddings and water boxes. However, due to much higher system thermal efficiency and higher fissile material conversion ratio of the core, the SCWR fuel utilization may be higher than that of LWRs.

The active fuel length is 4.2 m, which is a little longer than typical PWR fuels, to reduce the linear heat generation rate. Control rods are used for primary reactivity control, which are vertically inserted into and withdrawn from the core by the CRDMs mounted on the top of the RPV. To ensure adequate shut down margin and to minimize the local peaking during the entire operation cycle, the burnable poison Er_2O_3 is incorporated in the fuel.

In order to simplify structural design and obtain more uniform moderation, the standard fuel assembly cluster is composed of 4 square sub-assemblies as shown in Fig.4, and each of them consists of 56 fuel rods and a square water (moderator) rod in the center surrounded by a square channel box, and the cruciform control rods similar to that of BWRs are used. The fuel rods contain UO_2 pellets like PWR fuels in the modified stainless-steel cladding. In the heat-insulated water rod, lower temperature water flows downward to keep enough moderation in the core.

A 9×9 square arrangement for fuel rods in each subassembly is adopted, while central moderator box takes up 5×5 fuel rod cells. A dual wire is wrapped around each fuel rod, leaving a clearance of 0.1 mm between the wire and the fuel rods in order to mainly space the fuel rods and to enhance the heat transfer among the fuel rods. The fuel assembly structure material and fuel rod cladding material are 310S stainless steel, which shows good performance under supercritical water condition in the tests performed by NPIC and other researchers. The cross control rod with a width of 8mm is located at the center of 4 moderator box, which is inserted into the cross channel from the top of assembly.

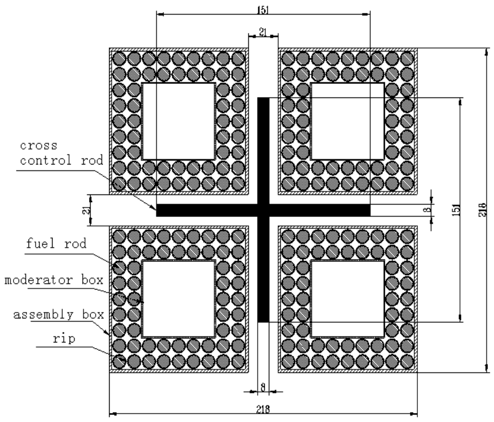


Fig.4 2×2 assembly cluster of CSR1000

The fuel assembly cluster has a total length of 5.9 m, which is composed by fuel rods, top/bottom nozzle, adaptor plates, assembly boxes, nozzle connectors, moderator boxes, moderator tubes, and etc（Fig.5）. In order to strengthen the laterally support and connect subassemblies into a cluster, 4 grids with a height of 30 mm are designed. Combining four sub-assemblies to form an assembly cluster allows reducing the number of individual control rod drives to similar numbers as in a PWR. A preliminary choice of mature fuel design of PWRs is adopted, with a diameter of 9.5 mm. In order to accommodate more fission gas and reduce the central temperature of fuel pellets, the holed fuel pellets are used.

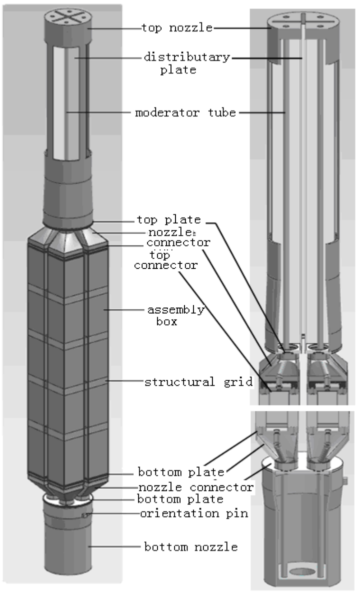


Fig. 5 Illustration of fuel assembly cluster

2.3. Fuel handling systems

To be designed referring to those of BWRs.

2.4. Primary circuit component description (e.g. vessels, internals, etc)

Considering the reactor structure feasibility, reliability and demands of physics and thermal -hydraulics, a two-pass reactor is proposed on which the CSR1000 general structure is designed. The reactor system mainly includes reactor vessel, reactor internals, fuel assemblies and control rod drive mechanisms (Fig.6).

Most water enters the upper plenum from the holes on the top of core-barrel and pressing-barrel, and then it's distributed as the 1^{st} pass coolant (38.0%), moderator in assemblies (both the 1^{st} and the 2^{nd} pass assemblies, 6.5% and 13.7% respectively) and moderator between assemblies (35.8%), then flows downward to the gathering structure in lower plenum. A small portion of water (6.0%) flows downward from the ring cavity between core-barrel and reactor vessel to the gathering structure. All the water collected in gathering structure then flows upward, working as the 2^{nd} pass coolant, after mixing in the steam plenum, supercritical coolant leaves the reactor through the thermal-sleeves in the outlets of the reactor vessel.

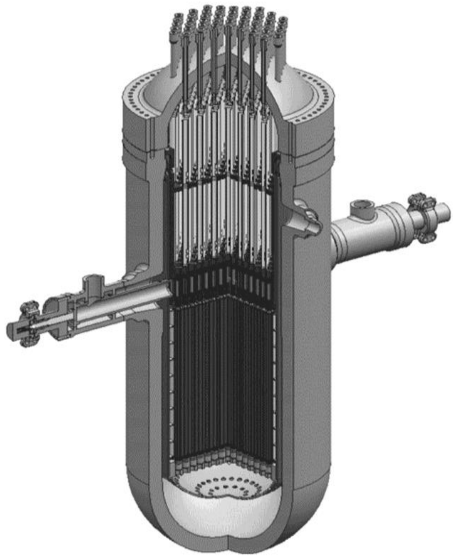


Fig.6 Reactor structure diagram

Reactor Vessel

The appearance of the reactor vessel is nearly the same with PWRs except the outlet structures for the interfaces between CSR1000 vessel and internals are similar to PWRs. And there are no penetrations on lower head of the vessel in order to enhance the safety and reliability.

Moreover, the two-pass design ensures that the temperature of the coolant touching the vessel inner wall does not exceed the allowable temperature limit of the existing RPV material. Thus the current 508-III steel will be usable if some proper improvements are made.

CSR1000 reactor vessel mainly consists of upper head component, cylinder component, lower head component, outlet structures, fasteners and sealings. The upper head component is a hemispherical head, and the control rod drive mechanism nozzles and penetrations for instrumentation are disposed on it. The thickness of flange is about 680mm. The inlet and outlet nozzles are placed on the cylinder component with an axial height difference. The lower head component is also a half-sphere head. The thermal-sleeves are designed with the outlet structures to divide supercritical coolant and the inner surface of vessel since the outlet coolant temperature is too high for the vessel material.

Reactor Internals

Besides the traditional reactor internals' function, CSR1000 reactor internals are also the key components for realizing the two-pass coolant flow. Simultaneously, internals divide the regions with large temperature difference and provide reliable sealings.

CSR1000 reactor internals are designed according to the core design and coolant flow demands, including lower internals, upper internals, pressing spring, thermal sleeves and gathering structure. Reactor internals hang on the reactor vessel flange support and are located by four locating slots.

Lower internals consist of core-barrel, lower core plate, reflectors and sealing rings. Core-barrel wall thickness is about 60mm.

The gathering structure is a half-ellipsoid with flow holes, together with lower core plate, forms a lower collecting plenum, which is an important part to realize two-pass coolant flow. In this collecting plenum, the 1st pass coolant and moderator is blended and redistributed.

Two sealing rings are “C”-shape rings, which sit on the support inside the core-barrel to realize the sealing between core-barrel and upper core plate, core-barrel and pressing-barrel separately.

Thermal sleeves are designed to separate the coolant for the outlet coolant temperature is over 500℃ and exceeds the reactor vessel material allowance (350℃).

Upper internals consist of pressing-barrel, upper core plate, lower support plate, upper baffles, upper support plate and guiding assemblies. Pressing-barrel is used to press fuel assemblies and its wall thickness is about 55mm.

The steam plenum is another key part for realizing two-pass coolant flow, formed by upper core plate, lower support plate, upper baffles and pressing-barrel. The steam plenum implements the following functions: distributing the downward flow, blending the 2nd-pass outlet coolant, providing reliable sealing and bearing large thermal-stress.

In CSR1000, cruciform control assemblies are adopted. However, the insertion and locating mode are different with the structure in BWR, so a new type guiding assembly with rail is designed.

Control Rod Drive Mechanisms

Compare to the widely used control rod drive mechanism (CRDM) in current PWRs, the CRDM working conditions in SCWR are more challenging. Therefore, according to the working temperature, pressure and lifting load, some improvement work is needed and that would bring some technical challenge on the CRDM wiring coil design.

Design characteristics of reactor structure

The steam plenum is above the core of CSR1000 for separation of the first flow coolant, the second flow coolant and moderator. The steam plenum supplies a space for collection of outlet coolant, and a flow channel for coolant runoff. The structure is very simple and convenient for installation.

The flow regulators are below the core of CSR1000 for flow distribution, located on internal of the lower core plate. They adapt the core power distribution by transferring position or changing another one on refuelling shutdown condition.

The thermal sleeves are installed between barrel and RPV of CSR1000 to supply a channel for the outlet supercritical coolant to protect PRV from large thermal stress. The structure is very simple and convenient to install and remove.

2.5. Auxiliary systems (e.g. heat removal systems, cooling systems etc)

The auxiliary systems in the nuclear island are the Normal Residual Heat Removal and low pressure injection system and many other auxiliary systems such as instrument and service air system, heating ventilating and air conditioning system.

The RNS diagram is shown in Fig.7. Other auxiliary systems will be designed in the following R&D stages.

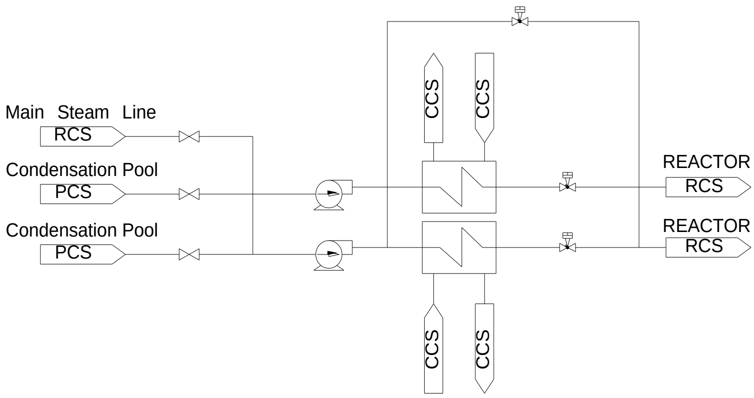


Fig. 7 Normal Residual Heat Removal and low pressure injection System diagram

2.6. Operating modes

(not provided)

2.7. Standard Fuel cycle (once through, closed, etc)

The standard fuel cycle for the CSR1000 will be a closed cycle so as to make full use of fissile material resources.

2.8. Alternative Fuel options

No information is provided now.

2.9. Spent nuclear fuel and disposal plans if any

No detailed information is provided now.

2.10. Examples of energy systems with NPPs of this kind, if any

No information is provided now.

3. DESCRIPTION OF SAFETY CONCEPT

3.1. Safety concept and design philosophy and licensing approach

The safety goal of the CSR1000 design to is to achieve significant reductions in the risks to the public, workers and the environment. As such, the design should provide significant improvements in the expected frequency of design basis accidents (DBA) and beyond design basis accidents (BDBA) as well as reductions in core damage frequency (CDF), large early release frequency (LERF), and safety system unavailability.

The design philosophy for safety and reliability are as follows:

- Maximum utilization of the matured, proven technologies that have been accumulated in the successful commercial operation of PWRs as well as supercritical pressure fossil-fired power plants (FPPs).
- Safety system development based on inherent feature of water-cooled reactor and well-developed PWR safety technologies. The inherent feature includes negative void (density) and Doppler coefficients. The well-developed PWR safety technologies mainly include reactivity control systems and emergency core cooling systems (ECCS).
- CSR1000 is once-though cooling system, without water level monitoring in the reactor vessel, the ECCS design is needed to achieve the fundamental safety requirement which is to ensure a sufficient core coolant flow rate ("coolant supply from clod-leg" and "coolant outlet at hot-leg") after reactor shutdown for decay heat removal.

It is envisioned that the licensing approach will be the similar to the processes used for traditional GEN-III LWRs.

3.2. Provision for simplicity and robustness of the design

Mentioned in the related sections above.

3.3. Active and passive systems as well as inherent safety features; indication of whether the system is the main or backup system

The Safety System of CSR1000 includes the following systems:

<u>High Pressure Injection System</u>

At the beginning of Loss of Main Feedwater (LOMF) event, the main system is at high pressure, enough water should be injected into the reactor vessel for cooling the core at this time. The high pressure injection is carried out by using the high pressure feedwater tank of the Reactor Coolant System (RCS). The relative diagram is shown in Fig.8.

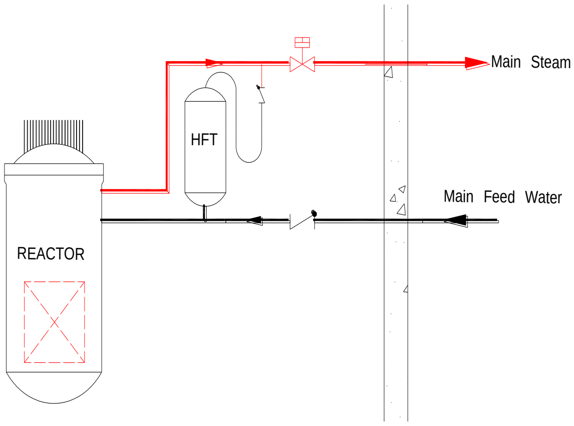


Fig.8　High Pressure Injection System diagram

ADS: Automatic Depressurization System

The Automatic Depressurization System is separated as ADS L1 and ADS L2. There is one isolation valve (normal closed) and one control valve (normal open) set on the ADS L1 line. The ADS L1 suction is connected to a main steam line and its downstream is in the condensation pool. When the RCS pressure rises up to 27MPa (abs), the ADS L1 isolation valve automatically open for discharging the super critical steam to the condensation pool.

There is an isolation valve (normal open) and one squib valve (normal closed) set on the ADS L2 line. The ADS L2 suction is connected to a main steam line and its downstream is directly to the atmosphere of the containment dry well. When the RCS pressure is rising up to 28MPa (abs), the ADS L2 squib valve automatically open for discharging more super critical steam to the atmosphere.

Normal Residual Heat Removal and low pressure injection System

See 2.5.

PXS: Passive Core Cooling System

The Passive Core Cooling System (PXS) is separated as the following subsystems:

- Passive residual heat removal subsystem (PRHR)
- Passive reactor injection subsystem.

The PRHR can remove the residual decay heat from reactor to the condensation pool by a passive heat exchanger (PRHR HX). The PRHR HX upstream is connected to the RCS main steam line via a normal open isolation valve, its downstream is directly connected to the reactor with a high pressure injection line. When the normal closed isolation valve which is set on the high pressure injection line open, the PRHR HX can automatically operate for heat transfer from reactor core to the condensation pool.

The passive reactor injection subsystem connects the condensation pool directly to the reactor with a passive reactor injection line. When the RCS depressurized to the atmospheric pressure after some accidents, the 2 isolation valves automatically open for providing enough water for cooling the reactor when the main system is depressurized to the normal pressure and the active core cooling systems are invalid.

PCS: Pressure Suppression Type Containment System

The CSR1000 containment is pressure suppression type (see Fig. 9). The containment is divided into areas: reactor pit, wet well, dry well. The reactor pit is used for containing the reactor vessel. The dry well is used for containing the RCS components and some safety system components. The wet well has a large condensation pool for suppressing the containment when the steam is discharged from the RCS to the containment.

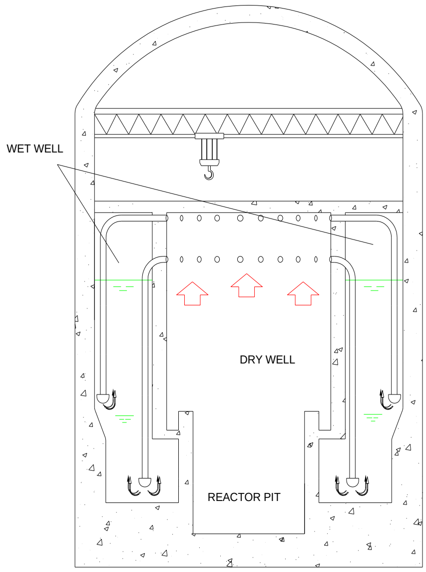


Fig.9 CSR1000 Suppression Type Containment

There are a lot of pressure balance pipes set in the containment. When the dry well is pressurized by the steam which is discharged from the main system, the steam will be discharged to the condensation pool and condensed via the balance pipes. In this way, the containment can be suppressed after the ADS operation or a LOCA.

PIS: Reactor Pit Injection System

The Reactor Pit is connected to the condensation pool by a balance line with a normal closed isolation valve. When the reactor core is melted and dropped in the vessel bottom, the PIS can inject enough water to the reactor pit for cooling the bottom surface of the reactor vessel.

Inherent safety-related features

As SCW is a single-phase fluid there is no DNB risk for CSR1000 and stainless steel fuel cladding reduces risks of H_2 explosion. Moreover, simplification of systems and components such as SGs, pumps etc. greatly reduce the possibilities of system and equipment failures causing abnormal conditions.

3.4. Defence in-depth description

The concept of defence in depth must be applied to all safety related activities to prevent accidents and mitigating the consequences of accidents. Application of the concept of defence in depth in the CSR1000 design provides several levels of defence including inherent features, equipment and systems, which were already considered in the CSR1000 conceptual design.

The CSR1000 inherent safety features described above are beneficial for the first three levels of defense. For the third level of defense, the CSR1000 applies the passive safety systems since the simplified primary circuit and large temperature difference between the reactor inlets and outlets are advantageous for the design of passive safety systems.

The fourth level of defence is to mitigate the consequences of accidents that result from failure of the third level of defence in depth. This part will be deliberated in the next R&D stage.

3.5. Safety goals (core damage frequency, large early release frequency and operator grace period)

The CSR1000 design objectives of the core damage frequency, large early release frequency and operator grace period are qualitatively defined in the conceptual design stage as follows:

- Core damage frequency: lower than that of the GEN-III
- Large early release frequency: lower than that of the GEN-III
- Operator grace period: longer than that of the GEN-III.
- Quantitative studies will be performed in the next R&D stage.

3.6. Safety systems to cope with Design Basis Accidents

Safety systems to cope with design basis accidents are described in the 3.3**.** A series of design basis accidents have been analysed using safety analysis code which developed for CSR10000. Loss of flow accident and loss of coolant accident are typical accidents among these DBA. The analysis results show that the safety system design of CSR1000 can effectively mitigate the DBA consequence and guarantee the reactor safety.

3.7. Safety systems to cope with Severe accidents (beyond design basis accidents)

To be addressed in the next R&D stage.

3.8. Provisions for safety under seismic conditions

To be addressed in the next R&D stage. Simplification of systems and components such as SGs, pumps, etc. greatly reduce the cost to seismically restrain key safety systems and equipment.

3.9. Probabilistic risk assessment

In the development of the CSR1000 design, probabilistic studies of Level 1 and Level 2 PSA will be performed to support and to optimize the design of the plant systems and processes in the next R&D stage. This practice will help achieve a well-balanced design and process and provides added assurance that the overall plant design will comply with the general safety objectives.

3.10. Emergency planning measures

To be addressed in the engineering design stage.

4. PROLIFERATION RESISTANCE)

To be addressed in the engineering design stage.

5. SAFETY AND SECURITY (PHYSICAL PROTECTION))

To be addressed in the engineering design stage.

6. PLANT PERFORMANCE

To be addressed in the next R&D stage.

7. DEVELOPMENT STATUS OF TECHNOLOGIES RELEVANT TO THE NPP

No information provided.

8. DEPLOYMENT STATUS AND PLANNED SCHEDULE)

8.1. Information on research and technology development status

The CSR1000 conceptual design and preliminary R&D work were completed in 2013. The preliminary design of a demonstration reactor is being pursued.

8.2. Companies/Institutions involved in RD&D and design

NPIC takes the overall responsibilities and undertakes main part of the CSR1000 deigns and R&D work in cooperation with several Chinese universities and research institutes.

An overall time frame within which the design could be implemented was issued by NPIC which aims at completing construction of the technical demonstration reactor for CSR1000 around 2027.

9. REFERENCES

http://www.npic.ac.cn

Appendix: Summarized Technical Data (CSR1000)

General plant data		
Reactor thermal output	2300	MWth
Power plant output, gross	～1000	MWe
Power plant efficiency, net	43.5	%
Mode of operation	baseload	
Plant design life	60	Years
Plant availability target	>90	%
Seismic design, SSE	0.3	g
Primary Coolant material	310S Stainless Steel	
Moderator material	Light water	
Thermodynamic Cycle	Rankine	
Type of Cycle	Direct	Direct/Indirect
Safety goals		
Core damage frequency	Lower than Gen-III	/RY
Large early release frequency	Lower than Gen-III	/RY
Operator Action Time	Longer than 72	hours
Nuclear steam supply system		
Steam flow rate at nominal conditions	1190	kg/s
Steam pressure/temperature	25/500	MPa(a)/℃
Feedwater flow rate at nominal conditions	1190	kg/s
Feedwater temperature	280	℃
Reactor coolant system		
Primary coolant flow rate	1190	kg/s
Reactor operating pressure	25	MPa(a)
Core coolant inlet temperature	280	℃
Core coolant outlet temperature	500	℃
Mean temperature rise across core	220	℃
Reactor core		
Active core height	4.2	m
Equivalent core diameter	3.379	m
Average linear heat rate	15.6	kW/m
Average core power density	61.1	MW/m^3
Fuel material	UO^2	
Cladding tube material	310S Stainless Steel	
Outer diameter of fuel rods	9.5	mm
Rod array of a fuel assembly	square	
Number of fuel assemblies	157	
Enrichment of reload fuel at equilibrium core	5.6 or 6.2	Wt%
Fuel cycle length	12 or 18	months
Average discharge burnup of fuel	～33000 or～45000	MWd/tU
Burnable absorber	Er_2O_3 or Gd_2O_3	

Control rod absorber material	B_4C	
Reactor pressure vessel		
Inner diameter of cylindrical shell	4855	mm
Wall thickness of cylindrical shell	440	mm
Total height(max)	15980	mm
Base material	508-III steel	
Design pressure/temperature	27.5/550	MPa(a)/℃
Steam generator (if applicable)	**N/A**	
Reactor coolant pump (if applicable)	**N/A**	
Pressurizer (if applicable)	**N/A**	
Primary containment		
Type	Pressure-suppression	
Overall form (spherical/cylindrical)	cylindrical	
Dimensions (diameter/height)	23/35	m
Design pressure/temperature	0.5/145	MPa(a)/℃
Design leakage rate	0.3	Vol%/day
Is secondary containment provided?	no	
Residual heat removal systems		
Active/passive systems	Active+passive	
Safety injection systems		
Active/passive systems	Active+passive	
Turbine		
Type of turbines	Impulse	
Number of turbine sections per unit (e.g. HP/MP/LP)	1HP(single flow) 1MP(double flow) 3LP(double flow)	
Turbine speed	3000	rpm
HP turbine inlet pressure/temperature	24.65/493	MPa(a)/℃
Generator		
Rated power	1176	MVA
Active power	1006	MW
Voltage	26	kV
Frequency	50	Hz
Condenser		
Type	single backpressure	
Condenser pressure	5.0	kPa(a)
Feedwater pumps		
Type	Steam-driven	
Number	2	
Head at rated conditions	2220	m
Flow at rated conditions	0.6	m^3/s
Pump speed	5770(max)	rpm