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ADVANCED PASSIVE PWR AC-600 — DEVELOPMENT ORIENTATION OF NUCLEAR POWER REACTORS IN CHINA FOR THE NEXT CENTURY

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Abstract

Based on Qinshan II Nuclear Power Plant that is designed and constructed by way of self-reliance, China has developed advanced passive PWR AC-600. The design concept of AC-600 not only takes the real situation of China into consideration, but also follows the developing trend of nuclear power in the world. The design of AC-600 has the following technical characteristics: Advanced reactor: 18~24 month fuel cycle, low neutron leakage, low power density of the core, no any penetration in the RPV below the level of the reactor coolant nozzles; Passive safety systems: passive emergency residual heat removal system, passive-active safety injection system, passive containment cooling system and main control room habitability system; System simplified and the number of components reduced; Digital I & C; Modular construction. AC-600 inherits the proven technology China has mastered and used in Qinshan II, and absorbs advanced international design concepts, but it also has a distinctive characteristic of bringing forth new ideas independently. It is suited to Chinese conditions and therefore is expected to become an orientation of nuclear power development by self-reliance in China for the next century.

1. INTRODUCTION

Around the beginning of next century, the development of nuclear power in the world will go into a new period of updating and upgrading. People universally follow with interest which type of advanced new reactors will be adopted in new generation of nuclear power plants. In many years, nuclear power suppliers in the world have been throwing in a great amount of man-power, material and financial resources to research into and develop, in accordance with utility requirements, a new generation of nuclear power reactors which is safer, more economic and more reliable. A lot of positive results of the research and development have been obtained. Among various reactor types that are geared to the needs of next generation of nuclear power plants, a revolutionary type of advanced PWR with passive safety systems has attracted the attention of many people because of its higher level of safety and economy.

On the basis of the nuclear power development policy of "taking ourselves as the dominant factor and co-operating with foreign countries", China has closely been following the trend and progress of international development of nuclear power from the 1980s, and launched research on new technology of next generation of nuclear power based on the national conditions. It is under this background that advanced passive PWR AC-600 has been developed on the basis of Qinshan II Nuclear Power Plant which is designed and constructed by way of self-reliance, AC-600 embodies the features of an advanced reactor: simplified systems, passive safety, digital I&C, and modular construction. Its safety and economy is improved. AC-600 inherits the proven technology China has mastered and used in Qinshan II, and absorbs advanced international design concepts, but it also has a distinctive characteristic of bringing forth new ideas independently. It is suited to Chinese national conditions and therefore is expected to become an orientation of nuclear power development by self-reliance in China for the next century.

Table I. Major Design Targets for AC-600

Parameter	Design Target
Construction cost	About 20% less than that of Qinshan II NPP
Core melt frequency	1×10^{-6}- to 1.5×10^{-6} /r-y
Availability factor	>85%
Refueling period	18 months
Construction period	4~5 years
Plant life time	60 years
Plant personnel exposure dose	0.5-1.0 man-Sv/year

The AC-600 has a large safety margin of operation because of the low power density of the reactor core. The high natural circulation cooling ability due to a small flow resistance of the primary system loop is very useful for reactor core decay heat removal during accidents. The AC-600 has a large reactor pressure vessel, a large pressurizer and a large water volume in the primary systems, which facilitate accident mitigation. The AC-600 design, which eliminates the high-head safety injection pumps, utilizes full pressure core makeup tanks and larger accumulators for the engineered safety features. The passive containment cooling system is used as the ultimate heat sink. All the measures mentioned above increase both the reliability and the capacity of the engineered safety very much, largely improving the safety of AC-600. The major design targets of AC-600 are given in Table 1.

The average linear power density of the AC-600 fuel rod is 13.78 kW/m, much smaller than that of Qinshan- II (16.087 kW/m). The small core power density makes for the reactor to have large thermal safety margins for normal operation and accident conditions.

The AC-600 design uses Gd_2O_3 burnable poison to reduce the excess reactivity of the reactor and the critical boron concentration. Because of the small critical boron concentration, a large negative temperature coefficient of reactivity can be obtained. The small excess reactivity and the large negative temperature coefficient of the core is one of the AC-600 design characteristics, largely improving the passive and inherent safety of the reactor to prevent power excursions induced by reactivity accidents.

The measures of elevating the vertical distance between the steam generators and the reactor core, and reducing the flow resistance, are used in the AC-600 design to increase the natural circulation cooling flow rate of the primary coolant. If the reactor operates at 25% of rated power, the natural circulation flow rate is 4852 t/h=1347.78 kg/s (15.12% of the rated flow rate) after the reactor coolant pumps shut down. The natural circulation flow rate increase is a very important part of the passive safety functions of the AC-600.

The passive emergency residual heat removal system on the secondary circuit side is mainly used in the event of station blackout, main steam line rupture or loss of feedwater supply. The system consists of an emergency feedwater tank, an emergency air cooler, valves and pipes for each loop. When station blackout occurs, the decay heat generated in the reactor core can be removed through use of natural circulation flow in the primary coolant system, in the secondary coolant system, and to the atmosphere, respectively.

In order to increase the reliability of the safety injection system, two full pressure core makeup tanks, two accumulators and four low-head safety injection/recirculation pumps, which are installed in the containment sumps, are utilized in the AC-600 design. In case of a large LOCA, the flow rate into the RCS from a core makeup tank is larger than that from a high-head safety injection pump in the conventional design. It is necessary for the AC-600 to use active pumps to perform the functions of the low-head safety injection/recirculation system.

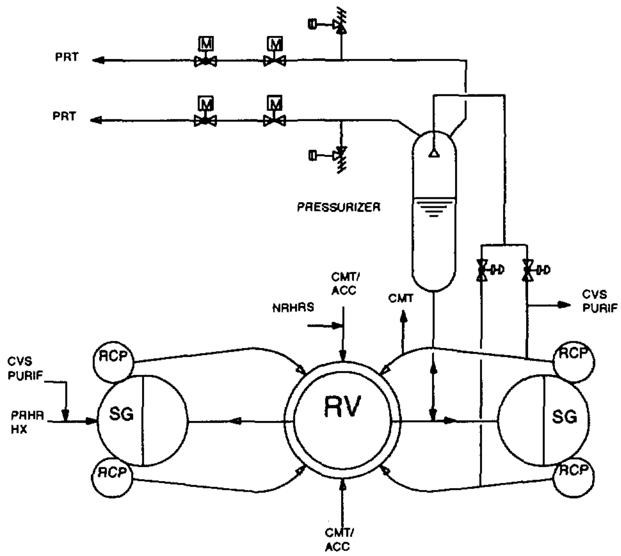

FIG. 1 Reactor coolant system flow diagram

The passive containment cooling system is used to remove the heat from the inside to the outside of the containment during a LOCA or a main steam line rupture inside the containment. First, the water in the tank on the top of the containment will be sprayed out to the outer surface of the steel shell of the containment by gravity, cooling the shell so as to decrease the pressure and the temperature inside the containment. After the tank on the top of the containment becomes empty, the natural circulation flow of air through the annulus between the steel shell and the concrete shell can remove the heat from the inside to the outside of the containment continuously. At the same time, the low-head safety injection/recirculation pumps, which are installed in the containment sumps, can return the borated water from the sumps into the reactor coolant system. The water absorbs the core decay heat and flows out through the break point (in LOCA condition).

2. DESCRIPTION OF THE NUCLEAR STEAM SUPPLY SYSTEMS

2.1 Primary circuit and its main characteristics

The AC-600 reactor plant is based on Qinshan II (2×600 MWe PWR NPP). But it is improved and enhanced in safety, reliability and economy as compared to Qinshan II. The major improvements include:

- utilization of advanced core design with low specific power, gadolinium in fuel, improved fuel management, arrangement of radial reflector and gray control assemblies;
- simplification of reactor coolant system by using high-inertia, high-reliability and low-maintenance hermetically sealed and canned motor pumps connected directly to the bottom of steam generators;
- utilization of passive safety systems to reduce the dependence on power supply;
- introduction of digital I&C systems emphasized on operability and maintainability, reliability and availability, and standardization and modularization;
- application of modular construction technology to save labor and time during construction and maintenance.

The primary circuit of the AC-600 uses two loops - with a steam generator and two reactor coolant pumps in a "one-hot-leg-two-cold-legs" arrangement - connected in parallel and symmetrically to the reactor, a pressurizer, and a relief tank. The schematic flow diagram of the reactor coolant

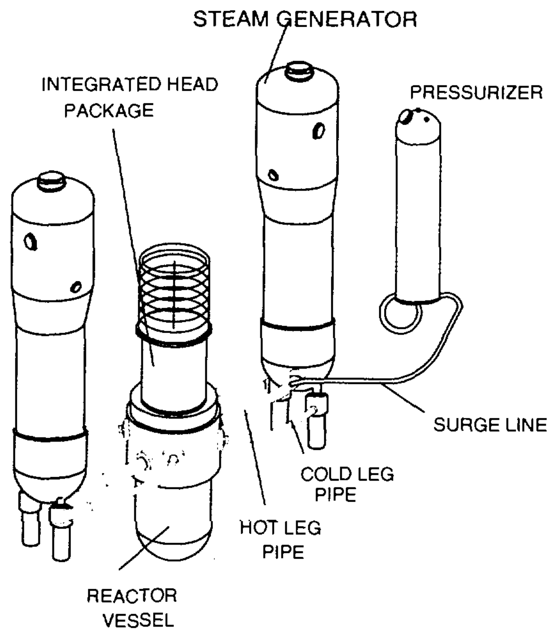

FIG. 2 Isometric view of the reactor coolant loops and major components

system and isometric view of the reactor coolant loops and major components are shown in Figures 1 and 2.

2.2 Reactor core and fuel design

The reactor core consists of 145 17×17 advanced fuel assemblies, 17 control rod assemblies and other fuel associated assemblies. There are 45 black rod (Ag-In-Gd) and 12 gray rod (stainless steel) assemblies in the core. The burnable poison (Gd_2O_3) is dispersed in the fuel. The average burn-up is 42000 MWd/tU.

The control rod drive mechanism (CRDM) for conventional PWR will be adopted in the design for the AC-600, except that wires to be used in the electromagnetic coils for the AC-600 CRDM are melting-extruded. The operating temperature of the coils will be higher than 300 °C (about 350 °C).

2.3 Primary components

2.3.1 Reactor Pressure Vessel

A schematic drawing of the reactor pressure vessel and internals is shown in Figure 3. The reactor vessel encloses all components of the reactor core. It is made of SA 508-3 steel made in China. Because of the lower power density in the core, the energy storage in fuel elements is less and the cladding temperature would be lower under accidental conditions. That results in increased thermal and safety margins. The nozzles of the CRDM and the in-core instrumentation are located on the closure head. There are no penetrations in the reactor pressure vessel below the level of the reactor coolant nozzles.

2.3.1.1 Reactor Internals

The reactor internals consist of two major assemblies—the lower internals and the upper internals. The main function of the reactor internals is to provide protection, alignment and support for the core and control rods to maintain safe and reliable reactor operation. In addition, the reactor internals help to direct the main coolant flow to and from the fuel assemblies, to support instrumentation within the reactor vessel and to provide protection for the reactor vessel against excessive radiation exposure from the core.

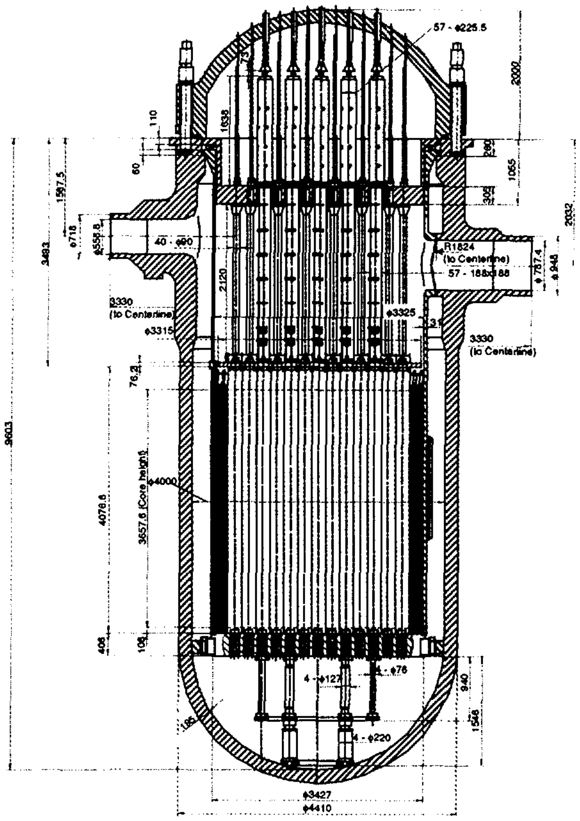

FIG. 3 Reactor pressure vessel and internals

The lower core support assembly consists of the core barrel, lower core support plate, secondary core support, vortex suppression plate, radial reflectors, radial supports, and related attachment hardware.

The upper core support assembly consists of the upper support, the upper core plate, the support columns, and the guide tube assemblies.

2.3.1.2 Steam Generators

The steam generator is of the vertical U-tube type. The material of the U-tubes is thermal-treated Inconel-690 (I-690TT). Two canned-motor pumps are welded to the steam generator bottom head. In this case, the U-shaped cross-over leg of the coolant pipe is eliminated.

2.3.1.3 Pressurizer

The pressurizer for AC-600 is of conventional design. It is about 30% larger than that normally used in a plant of comparable power rating, so as to increase transient operation margins.

2.3.1.4 Reactor Coolant Pumps

The reactor coolant pump (RCP) is of the mixed flow, canned-motor pump type. There are four canned-motor pumps connected to the two steam generator bottom heads directly. Lubrication and cooling of the RCP are performed with water. In order to ensure an adequate inertia of the canned-motor pump, a motor and pump design with a rotating inertia of 0.15 t-m^2 will be employed.

2.3.1.5 Primary Coolant Piping

Reactor coolant system piping is configured with two identical main coolant loops, each of which employs a single 787.4 mm inner diameter hot leg pipe to transport reactor coolant to a steam generator. The two reactor coolant pump suction nozzles are welded directly to the outlet nozzles on the bottom of the steam generator channel head. Two 558.8 mm inner diameter cold leg pipes in each loop (one per pump) transport reactor coolant back to the reactor vessel to complete the circuit.

2.4 Reactor auxiliary systems

The reactor auxiliary systems include chemical and volume control system, equipment cooling system, waste processing and draining system and fuel storage, transfer and handling system.

3. INSTRUMENTATION AND CONTROL SYSTEMS

3.1 Design concepts including control room

The monitoring and control system provides an automated diagnosis of the state and the operating conditions of the NPP. Monitoring and presentation of information on the reactor coolant system, on all the safety-related systems, on the containment, on all operation conditions of the NPP and on remote control of these systems is provided. A post-accident monitoring system is provided to estimate the state of NPP.

Facilities for presentation of information including displays for monitoring of safety systems ensure:

- indication of control rod position
- monitoring of neutron flux during operation, refueling and maintenance
- monitoring of level of radioactive contamination of the containment and the surrounded area
- preservation of adequate water level in the reactor vessel and the cooling systems
- scram (emergency protection) of the reactor
- protection of safety-related systems.

In case of a main control room failure, the reserve control room is to provide:

- reactor trip to hot shutdown condition
- maintaining of hot shutdown condition
- monitoring of sub-criticality
- putting into operation of confining systems
- reactor cooldown with some local operations.

Plant process control systems fulfil the automatic control of the following main controlled parameters:

- neutron flux in the core
- primary pressure
- secondary pressure
- water level in the steam generators
- water level in the pressurizer

The design value of the reactor neutron flux is maintained with the control bank of neutron absorbers, consisting of several rod cluster control assemblies, within ±2% of its nominal value.

The design value of the primary pressure is maintained by the pressurizer electric heaters and by spray valves on the pressurizer spray line from the reactor coolant pump exit side to the steam phase of the pressurizer within ±0.3 MPa.

The design value of the secondary pressure is maintained by an appropriate balance of reactor power and steam flow from the steam generators to the turbine or to the steam dumping devices within ±0.2 MPa.

The water level in the steam generators is maintained within ±180 mm of its nominal level by means of the steam generator feedwater supply controller actuating the control valve on the steam generator feedwater line.

The water level in the pressurizer is maintained within normal value by the level controller, actuating the control valves located on the make-up line, and make-up pumps.

During normal operation, the reactor's neutron power and the process parameters are maintained automatically by the reactor control system. Protection against transients due to the introduction of reactivity is ensured by the reactor protection system. When reaching the set-points of neutron flux or reactor period, the reactor protection system will warn the operators to take actions or will trip the reactor so that reactor safety can be ensured.

3.2 Reactor protection system and other safety systems

The list of automatic safety systems encompasses

- reactor scram (emergency protection) system;
- primary overpressure protection system;
- emergency core cooling system;
- system of passive residual heat removal from the secondary side of the steam generator;
- passive cooling system from the containment;
- system of quick-acting isolation valves in the main steam-lines;
- secondary overpressure protection system;
- diesel-generator system and
- system of reliable direct current power supply.

When any accidents occur, the reactor protection system is designed to perform reactor trip. Scram (emergency protection) is actuated by de-energizing the control rod drive mechanisms.

The following parameters provide input to the reactor protection system:

- increase of neutron flux
- decrease of reactor period
- dT/dt and dP/dt
- decrease of pressure in the reactor
- increase of pressure in the reactor
- decrease of the flow rate in the reactor
- decrease of water level in SG
- increase of water level in SG
- decrease of reactor coolant pump speed
- signal of safety injection.

The reactor protection system trips the reactor, meeting the design criteria under all design conditions.

To mitigate the consequences of ATWS, the protection actions will be taken to trip the turbine and to start the auxiliary feedwater system.

3.2.1.1 Reactor scram (emergency protection) system

The reactor scram (emergency protection) system provides reliable switch-off of the electric power supply to the rod drives, causing the scram (emergency shutdown) rods to drop into the core. In this case, a disappearing signal for the original cause does not stop the initiated scram (emergency protection) action.

3.2.1.2 Primary circuit overpressure protection system

The system comprises three identical pilot safety valve assemblies, which discharge steam or steam-water mixture from the steam phase of the pressurizer to the relief tank when the pressure in the pressurizer increases above the permissible one. The subsystem for receiving the steam or steam-water mixture involves a relief tank and pipelines connecting it with the outlets of the safety valves.

3.2.1.3 Emergency core cooling system

The emergency core cooling system (ECCS) comprises the following complex of subsystems initiated automatically:

- subsystem of core make-up tank with full pressure (high-pressure safety injection subsystem)
- subsystem of accumulator pressurized by nitrogen
- subsystem of low pressure active safety injection and recirculation

Except for the subsystems of low-pressure active safety injection and recirculation, sources of alternating current are not required for the fulfillment of ECCS functions. The air-operated valves, needed for the function of emergency heat removal, are driven by compressed air from the compressed air storage tanks. The power supply of the subsystems of low-pressure active safety injection and recirculation are provided by the diesel generators or by the offsite power source (during the recirculation stage after LOCA).

3.2.1.4 System of passive residual heat removal from the secondary side of the steam generator

The passive residual heat removal system removes the residual reactor power during a plant blackout with the aid of natural circulation on the secondary side of the steam generator (S.G.). It consists of two independent trains, each of them being connected to the respective reactor loop via the S.G. Each train has an emergency feedwater tank, a heat exchanger cooled by air and located outside the containment, and piping for steam and condensate circulation. The fail-open valves on the piping are driven by compressed air. The air-cooled heat exchanger rejects decay heat via the steam generators into the atmosphere outside the containment.

3.2.1.5 Passive cooling system from the containment

The passive containment cooling system removes heat from the containment in the event of loss-of-coolant-accident (LOCA) in the primary circuit. Steam released will condense on the inside of the containment shell that is cooled on the outside by natural circulating air and gravity drain of water from elevated tanks above the containment. The heat released to the inside of the containment is rejected to the atmosphere from the outer surface of the containment. The pressure of the atmosphere inside the containment is kept below the permissible design value.

3.2.1.6 System of quick-acting isolation valves in steam lines

The quick-acting isolation valves in the steam lines close at:

- water inventory in steam generators increases above the permissible level;
- increase of radioactivity in steam generators above the permissible level; and
- received signals of a steam line rupture.

The system provides for

- protection of the turbine from steam of high humidity;
- prevention of radioactivity release from steam generators; and
- restriction of steam blow-down during rupture of the secondary circuit.

3.2.1.7 Secondary circuit overpressure protection system

This system prevents the secondary circuit pressure to increase above the permissible level of 110% of secondary design pressure. It incorporates a power-operated relief valve and seven safety valves. These valves reject steam into the atmosphere.

4. ELECTRICAL SYSTEMS

4.1 Operational power supply systems

The normal and the emergency electric power supply system consists of two trains of 100% capacity, with each channel being divided into three groups considering reliability requirements and the time interval of loss of electric power.

4.2 Safety-related systems

4.2.1 Diesel generator system

Two physically separated diesel generators provide power supply to the safety-related systems, involving the recirculation pumps of the subsystem of low-pressure active recirculation.

Start-up of the two diesel-generators, one for each channel of reliable electric power and to be put into operation in the case of failure of main and reserve grid connections, is carried out in a time not exceeding 15 s from the moment of generation of a command to start-up.

4.2.2 System of reliable direct current power supply

This system consists of storage batteries. It provides the power supply to electromagnetic circuits for operating of safety systems and for recording of necessary post-accident parameters.

The D.C. electric power supply of the reactor control and protection system is ensured by batteries (in each train) designed for a discharge over 24 hours. Electric power supply from accumulator batteries during a station blackout situation is provided for the main control room and the auxiliary control room in full measure.

5. SAFETY CONCEPT

5.1 Safety requirements and design philosophy

The safety goals of nuclear power plants should include not only the protection of the environment and the public, but the protection of the plants themselves as well. The two sides of the safety goals can not be separated completely but are closely related to each other. It is quite evident that only under the prerequisite of the safety of the nuclear power plants themselves the goals of the environmental safety and the public health can really be achieved. Increasing the plant's own safety and preventing core melt should be emphasized so as to restore the public confidence in nuclear power.

Any intervention of operators during design basis accidents is prohibited for the first 30 minutes so that operators have enough time to consider the features of the accident occurred, which may prevent erroneous actions. Reactor safety actions are performed fully by the automatic control and protection system for the first 30 minutes.

In addition, provisions under design basis accidents are as follows:

- accident state monitoring such as in-core and sump level monitoring
- indication of control rod position, including lights and digits
- indication of radiation level and radioactive releases
- monitoring of the reactor safety shutdown states

The above systems are provided with automatic recording devices during any accidents. Alarm light signals and digital indications are also provided in the main control room.

5.1.1 Deterministic design basis

In the AC-600 design, the safety design basis depends on the Engineered Safety Features. Through deterministic safety studies, some key parameters should be optimized, such as the size of the CMTs, the size and the set-point of the cold leg accumulators, the size of the air coolers and the chimneys on the secondary side of the S.G.s, and the set-points for the control and protection system.

5.1.2 Risk reduction

PRA has played an important role in the AC-600 design. It was used to prove the design for features that could be improved from a risk standpoint. A number of plant design enhancements were made as a result of these PRA studies, and these have yielded a predicted core damage frequency of less than 10E-5/y and a significant release frequency of less than 10E-6/y, including:

- Passive safety-related systems reduce the dependence of safety-related system operation on electric power and compressed air. This significantly reduces the core damage frequency resulting from a loss of offsite power or station blackout event.
- The use of canned motor reactor coolant pumps eliminates pump seal loss of coolant accidents.
- Simplified passive safety-related systems reduce the need for, and importance of, operator action.
- There is no core damage sequence in which high-pressure failure of the reactor coolant system pressure boundary occurs.
- The analysis shows that many of the events, which, in the past, were leading contributors to the risk of nuclear power plants, are not as significant for the AC-600. Interfacing systems loss of coolant accidents, which are typically the highest risk for severe accident sequences, are virtually eliminated by the design of the AC-600.
- There is no containment failure due to hydrogen burns or other energetic phenomena. This is true even in events without water cooling on the outer surface of the containment shell.
- The integrated severe accident containment analysis shows that the AC-600 containment is capable of performing its function as the ultimate fission product barrier, and no containment failures occur from containment over-pressure or over-temperature. Thus, large release of steam

would result only from sequences involving containment bypass or failure of containment isolation.

- The frequency of significant release calculated in the PRA study provides a basis for eliminating the needs for an evacuation plan.

5.1.3 External and internal hazards

In the safety analysis, both external and internal hazards are taken into account. External hazards include natural phenomena such as earthquakes, wind, tornadoes, floods, and external missiles. Internal events are divided into the following five primary categories:

- manual shutdown,
- transients,
- loss of offsite power,
- loss of coolant accidents, and
- anticipated transients without scram.

5.2 Safety systems and features (active, passive, and inherent)

In the first stage of normal plant shutdown, the residual heat of the reactor and the coolant system is transferred to the secondary loop through the steam generators. The steam generated then enters the condenser through the turbine bypass system to be condensed. The auxiliary feedwater system supplies the steam generators with water. The whole process goes on till the pressure of the coolant system drops to 2.8 MPa and the temperature to 180 °C.

In the second stage of shutdown, residual heat removal is accomplished by the residual heat removal system. The residual heat removal system and the spent fuel pool cooling and purification system share the same equipment. It consists of two independent trains, each of which includes one pump and heat exchanger cooled by equipment cooling water. During normal plant operation, this system acts as the spent fuel pool cooling and purification system. During plant shutdown, one of its trains is used to transfer reactor residual heat. At the same time, the spent fuel pool is also cooled till the coolant pressure is under 0.1 MPa. Coolant temperature drops and will be kept at cold shutdown temperature.

During the plant shutdown and cooling process, the coolant pump is always in operation. It does not stop until the coolant temperature drops to 70 °C. Before it stops totally, coolant circulates in coolant loop. After that, coolant is driven by the spent fuel pit cooling pump.

Under the condition of a large leakage from the reactor coolant pressure boundary, emergency residual heat is removed by the emergency core cooling system and the passive containment cooling system. The water volume of the primary system is guaranteed by the emergency core cooling system in order to keep the fuel assemblies in the pressure vessel covered with water.

Under the condition of completely intact pressure boundary, the typical situation in which emergency residual heat removal is required is blackout. At this time, the passive heat removal system on the secondary side of the steam generators is automatically put into operation. Through natural circulation of primary coolant, natural circulation of secondary loop steam and condensed water, and natural convection of air in special ducts outside the containment, residual heat is removed to atmosphere. By this system the coolant temperature and pressure can be brought to the corresponding values for cold shutdown, or till the power supply is restored. Besides the condensed water from the air-cooled heat exchanger, secondary side system feedwater is also available from the emergency feedwater tank. So, the water volume is kept at the required value by natural circulation in the secondary system.

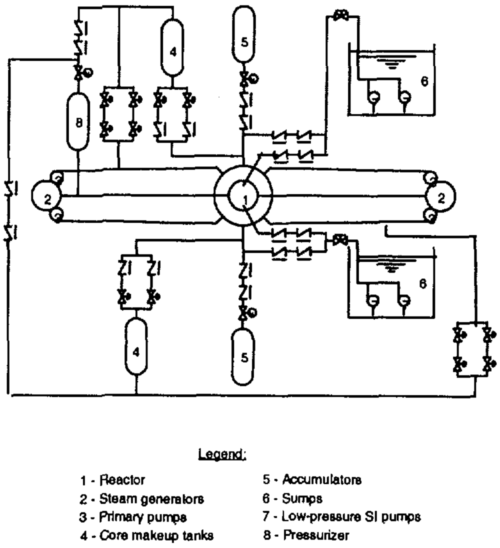

FIG. 4 Passive Safety Injection (PSI) system

5.2.1 Emergency core cooling system

The AC-600 utilizes an emergency core cooling system that is based on the principle of combining passive and active features. There are three subsystems for the emergency core cooling system. The AC-600 passive safety injection system is schematically shown in Figure 4.

The high-pressure injection subsystem consists of two reactor core makeup tanks. The middle pressure injection subsystem consists of two accumulators. The low-pressure injection and long term cooling subsystem consists of four low-pressure injection pumps taking suction from two special sumps in the containment. The low-pressure injection pump is of the vertical, low suction head type. The main functions of the emergency core cooling system are as follows:

- To supply water to the reactor in the event of abnormal leakage,
- In the event of LOCA, to inject water into the reactor and provide long term core cooling.

5.2.2 Passive residual heat removal system

The AC-600 passive residual heat removal system is illustrated in Figure 5. The function of this system is to remove the reactor core residual heat when the reactor loses its normal cooling resulting from station blackout or other accidents.

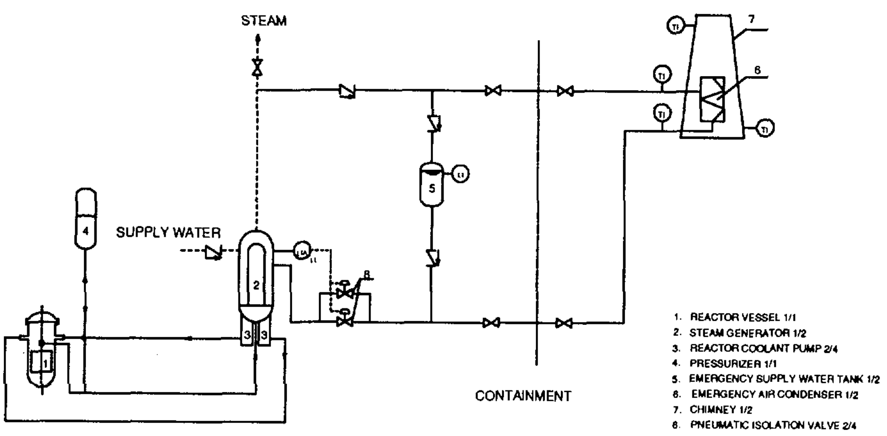

FIG. 5 Passive Residual Heat Removal (PRHR) system

This system has two trains. Each train consists of an emergency water tank and an emergency air-cooler. When blackout or other accidents occur, the isolation valves located at the outlet pipe of the emergency water tank are opened by a low-low water level signal for the steam generator, so that the emergency water tanks provide water to the secondary side of the steam generator by gravity and maintain the water level. The water in the steam generator absorbs the heat from the reactor coolant, when the water is heated into steam. The steam rises and passes through the emergency air-cooler where the steam is condensed into water. Simultaneously, the heat is transferred to the atmosphere. The condensed water returns to the steam generators by gravity, thereby a continuous natural circulation path will be established. Because of the cooling of the secondary side of the steam generators, a corresponding natural circulation in the reactor coolant system will also be established. In this way, the residual heat of the reactor core will be transferred to the atmosphere.

5.3 Severe accidents (Beyond design basis accidents)

5.3.1 Severe Accident Mitigation Strategy

The strategy applied to the AC-600 design to meet the severe accident challenge is a "defense-in-depth" strategy. The strategy includes the following steps:

- Planning accident management (AM) to prevent and mitigate severe accident consequences
- Prevention of high pressure core melt scenarios
- Protection of reactor pressure vessel by means of flooding of reactor cavity
- Protection of the containment integrity by means of control of hydrogen and control of pressure and temperature in containment
- Retention of radioactive material inside the containment and limitation of source terms

5.3.2 Severe Accident Prevention and Mitigation Features

5.3.2.1 Severe Accident Prevention Features

Following the accident initiator, the most important thing is to ensure inventory control and sufficient heat removal from the core. Failure to provide heat removal or inventory control results in core uncovery, fuel overheating and the potential for oxidation and melting of the reactor core.

In response to accident initiators identified through operating reactor experience and performance of probabilistic risk assessments, design features for the AC-600 to prevent the occurrence of such initiators from leading to a severe accident are still under study. These initiators include ATWS, mid-loop operation, station blackout, fire and intersystem loss-of-coolant accident.

5.3.2.2 Severe Accident Mitigation Features

External reactor vessel cooling is used to prevent reactor vessel failure and subsequent relocation of melted core debris into the containment. The engineered design features of the AC-600 promote flooding of the reactor cavity and submergence of the reactor vessel lower head.

Generation and combustion of large quantities of hydrogen can threaten containment integrity. So, in-vessel and ex-vessel hydrogen generation must be considered. The AC-600 is equipped with a hydrogen ignition system composed of thermal igniters. The intent of the system is to ignite the hydrogen as soon as sufficient hydrogen has accumulated to achieve a combustible mixture for detonation prevention.

One important boundary condition for severe accident control is that core-concrete interaction (CCI) with its potential for basemat penetration is prevented and hydrogen generation is limited. Many features in the AC-600 design to help mitigate the effects of CCI are still under development.

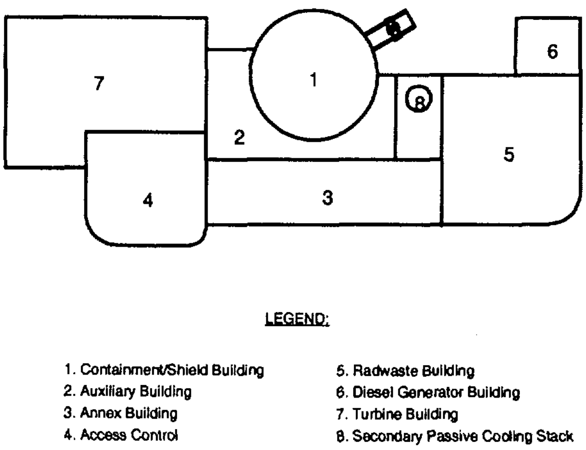

FIG. 6 General plant layout arrangement

High-pressure core melt ejection (HPME) and subsequent direct containment heating (DCI) can lead to early containment failure with large radioactive release to the environment. The AC-600 design provides a reliable depressurization system and cavity design features to decrease the amount of ejected core debris that reaches the upper containment.

Steam explosion resulting from fuel-coolant interaction is energetic and the rapid energy release can challenge the containment function. Steam explosion can occur either inside or outside the reactor vessel. On the basis of present knowledge, steam explosion has only a very low probability of occurrence. We expect that further work will confirm this. In any case, the risk of a steam explosion could be minimized by technical means.

The pressure and temperature in the containment can be controlled by means of the passive containment cooling system and filtered venting provided by the AC-600 design.

The equipment needed to perform mitigative function and the environmental conditions under which the equipment must function are under consideration to assure the equipment's survivability in severe accident.

In order to prevent the core from melting or radioactive release from the plant to the environment, operators are required to utilize all reasonable measures according to procedure H or U.

The fission chain reaction in the core should be stopped during severe accidents and the reactor returned to a controllable state. The measures of mitigation and accident management are researched through use of severe accident analysis.

6. PLANT LAYOUT

6.1 Buildings and structures, including plot plan

The plant layout is composed of seven principal building structures: the nuclear island, the turbine building, the annex building, the access control building, the secondary passive cooling stack, the diesel building and the radwaste building, as shown in Figure 6.

The nuclear island consists of a free-standing steel containment building, a concrete shield building, and an auxiliary building. The foundation for the nuclear island is an integral basemat, which supports these buildings. All safety-related equipment designed to perform accident mitigation functions is located in the nuclear island. The containment building houses the reactor coolant system and other related systems. The auxiliary building contains the main control room, instrumentation and control systems, class 1E direct current system, fuel handing area, mechanical equipment areas, containment penetration areas and main steam and feedwater isolation valve compartment.

The turbine building houses the main turbine, generator and associated fluid and electrical systems.

The annex building includes the health physics facilities, the non-1E AC and DC electric power systems, the lunch and cooker rooms, the technical support center and various heat, ventilation and air conditioning systems, the hot machine shop and access way to the upper and lower equipment hatches of the containment building for personnel access during outages.

The access control building includes the health physics facilities for the control of entry to and exit from the radiological control area, access ways for personnel and equipment to the clean areas of the nuclear island in the auxiliary building and to the radiological control area via the annex building.

The secondary passive cooling stack houses two air coolers. It is located on the top of the fuel handling area.

The diesel building houses two diesel generators and their associated heating, ventilation and air conditioning equipment.

The radwaste building contains facilities for the handling and storage of plant waste.

6.2 Reactor building

The reactor building consists of containment building and shielding building.

The containment building is a free-standing cylindrical steel containment vessel with elliptical upper and lower heads, it houses the reactor coolant system and other related systems.

The shield building surrounds the containment building. It is a reinforced concrete structure, with passive containment cooling system air baffle, passive water storage tank, and passive air diffuser. The shield building provides the protection for the containment building from external events and the required shielding for the internal postulated design basis accidents, it is also an integral part of the passive containment cooling system.

6.3 Containment

The containment is a free-standing cylindrical steel containment vessel with elliptical upper and lower heads. It houses the reactor coolant system and other related systems. The containment vessel includes the shell, hoop stiffener and crane girder, equipment hatches, personnel airlocks, penetration assemblies, and miscellaneous appurtenances and attachment. There are two floor elevations (grade access maintenance floor and operating deck) and a few lower equipment compartments within the containment building. Floor gratings are provided for access to equipment at other elevations. The passive core cooling system is located in the containment building. A seismically analyzed polar crane is provided in the containment and its bridge is sized for lifting the steam generator. The polar crane support is attached to the steel cylindrical shell of the containment. Two containment re-circulation heating, ventilation and air conditioning modules are provided. The chemical and volume control system equipment is located in the containment below the grade access maintenance floor level, a refueling machine and a fuel transfer system are designed in the containment.

The containment building provides shielding for the reactor core and the reactor coolant system during normal operations and provides a high degree of leak-tightness; it also is an integral part of the passive containment cooling system. The containment vessel and the passive containment cooling system are designed to remove sufficient energy from the containment to prevent the containment from exceeding its design pressure following postulated design basis accidents.

6.4 Turbine building

The turbine building for AC-600 houses the main turbine, generator, condenser, feedwater pump, deaerator and water storage tank, heater and associated fluid and electrical systems. It provides weather protection for the lay-down and maintenance of major turbine/generator components. The turbine building also houses the makeup purification system. No safety-related equipment is located in the turbine building. The turbine building consists of the main house, deaerator room and auxiliary area for condensate polishing.

The turbine building is 100 m long and 57 m wide. A bridge crane is provided for turbine maintenance.

The turbine building is a steel column and beam structure. The turbine building ground floor is a reinforced concrete slab. The foundation for the entire building is a reinforced concrete mat.

6.5 Service buildings

The service building for AC-600 include the protection auxiliary buildings, the buildings in the front area of the plant, the buildings in the depository area, the buildings adjacent to the site, the environment monitoring works, and BOP works.

7. TECHNICAL DATA

7.1 General plant data

Power plant output, gross	636	MWe
Power plant output, net	600	MWe
Reactor thermal output	1930	MWt
Power plant efficiency, net	31.1	%
Cooling water temperature	18	°C

7.2 Nuclear steam supply system

Number of coolant loops	2	
Primary circuit volume, including pressurizer	219	m^3
Steam flow rate at nominal conditions [1951 t/h]	2168	kg/s
Steam temperature/pressure	282.94/6.65	°C/MPa
Feedwater temperature		°C

7.3 Reactor coolant system

Primary coolant flow rate [13.49 m^3/s]	10002	kg/s
Reactor operating pressure	15.5	MPa
Coolant inlet temperature, at RPV inlet	292.8	°C
Coolant outlet temperature, at RPV outlet	327.2	°C
Mean temperature rise across core	36.8	°C

7.4 Reactor core

Active core height	3.658	m
Equivalent core diameter	2.922	m
Heat transfer surface in the core	6222.7	m^2
Total fuel weight (UO_2)	66.8	t
Average linear heat rate	13.78	kW/m
Average fuel power density	32.9	kW/kg U
Average core power density (volumetric)	78.69	kW/l
Thermal heat flux, Fq	311.1	kW/m^2
Fuel material	Sintered UO_2	
Fuel assembly total length	4100	mm
Rod array	Square, 17×17-25 (AFA)	
Number of fuel assemblies	145	
Number of fuel rods/assembly	264	
Number of guide tubes for control rods/instr.	24/1	
Number of spacers	8	
Enrichment (range) of first core	2.0, 2.5, 3.0	Wt%
Enrichment of reload fuel at equilibrium core	3.6	Wt%
Operating cycle length (fuel cycle length)	18	months
Average discharge burnup of fuel	42000	MWd/t
Cladding tube material	Zr-4	
Cladding tube wall thickness	0.57	mm
Outer diameter of fuel rods	9.5	mm
Overall weight of assembly	663.2	kg
Active length of fuel rods	3658	mm
Burnable absorber, strategy/material	Gd_2O_3,	Mixed with fuel
Number of control rod assemblies	57 (45 black & 12 gray rods)	
Absorber rods per control assembly	20	
Absorber material , black/gray rods	Ag-In-Cd/stainless steel	
Drive mechanism	Magnetic jack	
Positioning rate	72	Steps/min
Soluble neutron absorber	Boron acid	

7.5 Reactor pressure vessel

Cylindrical shell inner diameter	4000	mm
Wall thickness of cylindrical shell	205	mm
Total height	12220	mm
Shell and head material	Low alloy steel A508-III	
Design pressure/temperature	17.2/343	MPa/°C
Transport weight (lower part)[incl. Head]	440	t

7.6 Steam generator

Type	Vertical, U-tube heat exchanger	
Number	2	
Heat transfer surface area	5631	m^2
Number of U-type tubes	4640	
Tube dimensions	19.05×.1.09	mm
Maximum outer diameter	4843	mm
Total height	21000	mm
Transport weight	~350	t
Shell and tube sheet material	A508-III	
Tube material	I-690TT	

7.7 Reactor coolant pump

Type	Single-stage centrifugal pump with canned motor	
Number	4	
Design pressure/temperature	17.2/343	MPa/°C
Design flow rate (at operating conditions) [$3.374m^3/s$]	2500	kg/s
Pump head	72	m
Power demand at coupling, cold/hot	3340/2545	kW
Pump casing material	Stainless steel	
Pump speed	1488	rpm

7.8 Pressurizer

Total volume	36	m^3
Steam volume: full power/zero power	14.4/23.6	m^3
Design pressure/temperature	17.2/360	MPa/°C
Heating power of the heater rods	1440	kW
Number of heater rods	60	
Inner diameter	2100	mm
Total height	11760	mm
Material	A508-III	
Transport weight	~95	t

7.9 Pressuriser relief tank

Total volume	37	m^3
Design pressure/temperature	0.7/170	MPa/°C
Inner diameter(vessel)	3000	mm
Total height	5270	mm
Material	Austenitic stainless steel	
Transport weight	~12	t

7.10 Primary containment

Type	Dry, double wall, in steel/concrete	
Overall form (spherical/cyl.)	Cylindrical	
Dimensions (diameter/height)	37/57	m
Free volume	41500	m^3
Design pressure/temperature (DBEs)	430/130	kPa/°C
Design leakage rate	<0.25	vol%/day
Is secondary containment provided?	Yes, space between the walls	

8. FOUNDATIONS AND CONDITIONS FOR AC-600 DEVELOPMENT IN CHINA

8.1 Experimental and research base for nuclear power development

In order to fulfil self-reliance in nuclear power design, the Chinese government devotes much attention to research and development, and has thrown in large funds to construct a set of large test installations and research facilities at the Nuclear Power Institute of China (NPIC), which is located in Chengdu, Sichuam province of China. These installations and facilities are necessary for the AC-600 development. Part of them are listed below.

(1) Test facility for AC-600 passive emergency residual heat removal through secondary side of SG.

(2) AC-600 full pressure core makeup water test facility.

(3) Overall reactor core hydraulic simulation test facility.

(4) High neutron flux engineering test reactor.

(5) Comprehensive test facility of power device (a large loop and a small loop).

(6) Hydraulic test facility for control rod drive alignment under cold conditions.

(7) Comprehensive behavior test facility under hot condition for control rod drive alignment.

(8) Water chemistry test facility.

(9) Large scale thermal test facility.

(10) Well (shaft) used for seismic test and multi-excitation test facility.

8.2 Feedback of Qinshan II engineering experience

AC-600 is developed on the basis of Qinshan II, and its major parameters and main equipment are basically the same as those of Qinshan II, and therefore, a large amount of test and research result for Qinshan II can be used in the development and research of AC-600. The following summarizes some of the tests and research that have been performed for Qinshan II.

(1) Single and overall tests of reactor hydraulic simulations

(2) Control rod drive mechanism (CRDM) test

(3) Reactor protection system development and test

(4) Flow-induced vibration test for reactor internals

(5) Irradiation test of 4×4 fuel assembly in the high flux test reactor

(6) Test and research for reactor control and measurement systems, rod control system, and rod position indicating system

(7) Seismic test for control rod drive alignment

8.3 AC-600 R&D status

Starting from the eighth five-year plan, China has formally brought AC-600 development into line with the key state scientific research plan, and has performed a large amount of work on test and research, and made a lot of progress and breakthroughs. Table 2 lists major test and research items for AC-600 that have been completed. Through these tests and research, the passive safety concept of AC-600 has been verified in principle. The results obtained from these tests confirm the AC-600 design. All of this lays a good foundation for the future work.

TABLE 2. THE COMPLETED EXPERIMENTAL AND RESEARCH ITEMS OF AC-600

Designation	Contents	Finish time
Critical heat flux (CHF) test at low flow rate	The coolant flow per unit area of AC-600 core is relatively low. Under this low-flow-rate condition, measuring test data of departure from nucleate boiling on element surface and draw up a formula.	In 1993
Make-up test for make-up tanks at full pressure	Research on the passive characteristics of the make-up by a makeup tank at full pressure.	In 1997
Wind tunnel test for passive containment cooling system	Research on the correlation between flow resistance, flow duct shape and flow rate of the passive containment cooling system, and research on natural convection cooling characteristics.	In 1994
Test for the passive emergency core residual heat removal system on the secondary side	Research on the capability of the emergency core residual heat removal system on the secondary side, on the flow characteristic of the natural circulation and on the supporting means	In 1995
Flow characteristics test of the baffle plate in the exit plenum of SG	Research on the flow characteristics of the baffle plate in the exit plenum of the AC-600 SG and the entrance plenum of the main pump	In 1995
Digital I&C systems	Research on the control, adjustment and in-core measurement (Pneumatically-driven balls measurement)	In 1995

China is now working out future test and research plans based on what has already been achieved. It mainly includes the following items:

(1) New computer code development in order to be able to make calculations for Gd_2O_3 burnable poison and design for low neutron leakage. Performing zero-power physical test for the Ferro-water reflector in order to verify and improve computer code.

(2) Further improving reactor thermal-hydraulic design, and methodology for accident analysis and protection set-point determination.

(3) Developing computer codes used for passive safety system analysis.

(4) Carrying out digital I&C equipment development stage by stage.

(5) The general test scheme of passive containment cooling is now under proving. And the test facility is under preliminary design.

(6) Comprehensive test and research on passive core residual heat removal system.

(7) Further research on behavior of high performance fuel element and burnable poison rod.

9. CONCLUSION

The AC-600 is developed on the basis of the proven technology that China has mastered and used in Qinshan II NPP. Its design has fully absorbed ideas of advanced design concepts in the world and has a distinctive characteristic of bringing forth new ideas. The safety and economy have been improved. Technically, AC-600 is an advanced design that meets the requirements of nuclear power for the next century. It is suited to the real conditions of China. The self-reliance of nuclear power in China can be realized by means of the AC-600 development. Therefore, AC-600 represents the orientation of self-reliant nuclear power development in China for the next century.