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Review:

Generation III pressurized water reactors and China's nuclear power*

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Abstract: The design philosophy, overall performance, safety, and economy of three typical generation III (GIII) pressurized water reactors, EPR, AES2006, and CAP1400, are analyzed comprehensively in this paper. Based on comparison with and the lessons learned from the Fukushima nuclear accident, we forecast a future reactor for China's commercial nuclear power plant. Moreover, we put forward important technological fields of GIII nuclear power plants to which attention should be paid, including the enhancement of defense in depth, defense against extreme external events, severe accident mitigation, design simplification and standardization, improvement in economic competitiveness, load following capability, and adaptation to climate change.

Key words: Generation III (GIII) pressurized water reactor (PWR), Performance, Safety, Economy, Nuclear power
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1 Introduction

In 1957, Westinghouse successfully constructed the first commercial pressurized water reactor (PWR) plant, with a power output of 60 MWe, in Shippingport, USA. To date, nuclear power plants have become one of the most important options to optimize the energy portfolio, decrease carbon emissions, and alleviate air pollution. Over 14 000 years of reactor operation experience has been accumulated. According to the statistics from International Atomic Energy Agency (IAEA) (IAEA, 2015), the number of reactors in operation is 443; the total net installed capacity is 381.18 GWe; 65 plants are under construction, most of which are generation II (GII) or GII plus.

The evolution of fission reactor technology has been developed over four generations (GIF, 2014). The early prototype reactor proved the feasibility of using nuclear energy to generate electricity. Due to the oil crisis, most of the GII nuclear power plants were constructed during the 1970s and 1980s, which demonstrates the economic competitiveness of nuclear power plants when compared with other base-load electricity generation methods. Generation III (GIII) PWR is defined as technology that is in agreement with the requirement of the USA "utility requirement document" (URD) (EPRI, 2013) or the "European utility requirements document" (EUR) (EUR Organization, 2001), which put forward higher requirements for safety, economy, and provenness of technology for advanced light water reactors. Some GIII PWRs are under construction, such as EPR, AES2006, and AP1000. According to the forecast of generation IV (GIV) International forum, GIII nuclear power plants will be the main trend for commercial operation before 2070, with the GII plants being gradually decommissioned. GIV reactors focus

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on higher safety, nuclear non-proliferation, sustainable development, economic competitiveness, and versatile functions, which include gas-cooled reactors, lead-cooled reactors, molten salt reactors, sodium-cooled reactors, supercritical water reactors, and very high temperature reactors. It is very difficult for them to come into commercial operation due to the existing technology challenges before 2050.

From the perspective of technology approaches to improving nuclear power plant safety, three typical PWRs exist, including: EPR, by increasing the redundancy and diversity of active safety systems; AES2006, by using active plus passive safety systems and increasing the redundancy of active safety systems; and AP1000 and CAP1400, by adopting the philosophy of passive safety features and design simplification. In this paper, the design philosophy, overall performance, safety, and economy of three typical GIII PWRs will be analyzed comprehensively, from which, along with the lessons learned from the Japanese Fukushima nuclear accident, the main trend of reactor design for the development of China's nuclear power will be forecasted. Moreover, the potential development of technology fields of GIII nuclear power is also proposed.

2 Technology characteristics of typical GIII pressurized reactors

2.1 EPR

2.1.1 Overview

Based on experience from the French N4 and German KONVOI nuclear power plants, EPR is an evolutionary design developed by Areva to satisfy the requirement issued by the France nuclear safety regulator, Nuclear Safety Authority, of further improving the safety performance of nuclear power plants during the design phase. As a GIII PWR, it is in agreement with the requirements of EUR and the standards issued by IAEA. To enhance its economic competitiveness, the power output is increased. Its design objectives include: to decrease the core damage frequency (CDF) and large release frequency (LRF) by increasing the redundancy and physical separation of systems; to mitigate severe accidents; to optimize human–machine interfaces (digitalized instrument and control system adopted); to improve plant operation performance. EPR is a four-loop PWR, with one steam generator and a coolant pump for each loop, as shown in Fig. 1. More details and philosophy of ERP system organization are discussed by Bonhomme (1999). The emergency diesel generator building is separated from the reactor building to enhance physical separation, which is different from the traditional design (Areva, 2014).

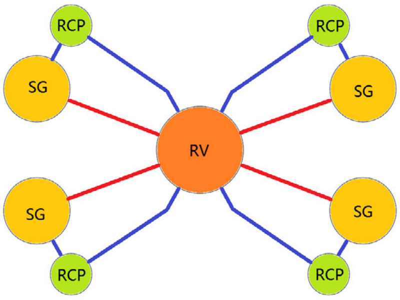

Fig. 1 Reactor coolant system of EPR

RV is the reactor pressure vessel; RCP is the reactor coolant pump; SG is the steam generator

2.1.2 Overall performance

The design life of EPR is 60 years, the thermal output is 4590 MW and the gross electrical output is about 1770 MW. To improve the plant's thermal efficiency, the heat transferring area of the steam generator is increased and an axial economizer is installed for the steam generator; moreover, the turbine-generator is also optimized. The steam pressure exiting from the steam generator is increased to 7.72 MPa. The axial economizer enhances the heat exchange efficiency between the primary side and the secondary side and increases the outlet steam pressure by 0.35 MPa as compared with a boiler-type SG with the same tube surface. The unit availability is greater than 92%. There are 241 fuel assemblies. A high burnup and low neutron leakage reactor core fuel management strategy is used for EPR. The fuel cycle length is extended to 24 months by increasing the fuel burnup. The reactor core is capable of loading 50% mixed oxide fuel (MOX) fuel, and 100% MOX would also be feasible if modifications of the reactor core were carried out. The duration of construction from the first concrete date to connection to the grid is less than 54 months. The safe shutdown earthquake is 0.25*g*. It is capable of load following operation and can be operated flexibly.

2.1.3 Safety

Several technical options are adopted for EPR to enhance the principle of defense in depth and improve its safety levels, which include: increasing the redundancy, diversity and physical separation, adopting core catcher, resistance to commercial airplane crashes, and layout optimization. The safety systems are designed with four trains (*N*+3), including a safety injection system, an emergency feedwater system, main steam line relief, a decay heat removal system (actuated when the steam generator is not able to remove decay heat), a component cooling water system, a service water system, and emergency electrical power, which is useful for preventive maintenance and repair work during normal operation. Design basis accident analysis proves that EPR satisfies the requirement that the operation action time is less than 30 min after a postulated accident.

One of EPR's design objectives is to limit off-site emergency response action to the vicinity of the site, which is indispensable to maintain the integrity of containment. It has established systematic severe accident mitigation strategy (Bouteille *et al.*, 2006). For the mitigation of severe accidents, a core catcher is used to confine melt debris within containment to prevent base mat melt (Fig. 2), which will avoid molten core-concrete interaction (MCCI) (Fischer, 2004; Mayousse, 2013). If reactor core debris penetrates the reactor pressure vessel, it will enter into the spreading area, which spreads the corium in a compartment provided with a protective layer and a special cooling system. The water in the in-containment refueling water storage tank will flow into the compartment and passively submerge the corium (Steinwarz *et al.*, 2001; Wittmaack, 2002; Fischer *et al.*, 2005). To keep the pressure within the containment below the design limit, the active containment heat removal system, which will be cooled by two separate dedicated trains of cooling water system and service water system, is used to cool the containment. Long-term cooling of the containment is provided by a containment spraying system.

Dedicated primary depressurization valves and a highly reliable decay heat removal system are installed to prevent core melting at high system pressures, which eliminates the risk of high-pressure failure of the reactor pressure vessel (RPV) and the risk of direct containment heating, which may potentially lead to early containment failure. Passive hydrogen recombiners and the design of connected compartments are adopted to control hydrogen concentration in order to avoid explosions. Ex-vessel steam explosions are prevented by minimizing water in the spreading area. Double-envelope containments are used to protect against airplane crashes and the potential subsequent release of radioactive substances (Bittermann *et al.*, 2001).

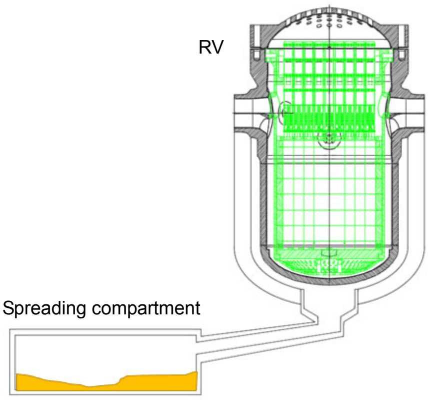

Fig. 2 EPR core catcher

2.1.4 Economy

Due to an increase in system redundancy, initial investment and operation and maintenance costs will increase for EPR. Therefore, to strengthen its economic competitiveness, positive factors for increasing the generation of electricity are improved, including unit net electrical output, thermal efficiency, plant design life, availability, and the extended fuel cycle length. Negative factors that would increase the cost of nuclear power plants are diminished, such as duration of construction.

2.2 AES2006

2.2.1 Overview

AES2006 is an evolutionary PWR developed by the Russian company Gidropress with reference to AES91 at the Chinese Tianwan nuclear power plant (NPP). AES2006 is a four-loop PWR, with one horizontal steam generator and a coolant pump for each loop (Kolchinsky, 2013), which is similar to that of EPR. However, the steam generator is horizontal which is different from other GIII power plant. Four passive emergency core cooling system (ECCS) accumulators are designed to provide safety injection for design basis accidents (Fig. 3).

The design of AES2006 is based on Russian regulatory documents, with consideration of the requirements of IAEA and EUR. AES2006 has a strategy for the prevention of severe accidents and a mitigation strategy, and also focuses on enhancing economic competitiveness. The design principles include: using verified technology to an extreme extent, adopting feedback from the Tianwan NPP, minimizing construction costs and duration, and considering the extended design conditions to ensure that the required safety levels of the nuclear power plant are met (Rosatom, 2014).

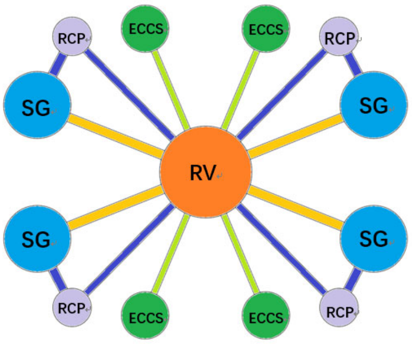

Fig. 3 Reactor coolant system of AES2006

2.2.2 Overall performance

Compared with AES91, the design life of AES2006 has been extended from 40 to 60 years, the power electrical output has been increased to 1197 MWe, and the thermal efficiency has been improved to 37% (gross). The steam pressure at the steam generator exit has been increased to 7.00 MPa. Analysis indicates that the availability is over 90%. There are 163 fuel assemblies. The fuel cycle length is 12 months, and it can be extended to 18–24 months. The reactor core is capable of loading MOX fuel. The average burnup of fuel is 60000 MWd/tU. The duration of construction from the first concrete date to connection to the grid is less than 54 months. The safe shutdown earthquake is 0.25*g*. It is capable of load following operation.

2.2.3 Safety

Except for defense in depth and three physical barriers, the design of AES2006 complies with the concepts of single failure criteria, redundancy, diversity, physical separation, inherent safety, and so on. Active and passive designs are adopted to withstand design basis accidents in terms of safety injection and residual heat removal, similar to VVER1000 (Mousavian *et al.*, 2004). The safety systems are designed with the principle of enhanced diversity and redundancy, which is realized by N+3. The safety systems are designed with four trains, which are arranged in four separate compartments to allow for physical separation (Kolchinsky, 2013). In the previous design, only normal operation, anticipated operational occurrences, and design basis accidents are considered. In addition to these, AES2006 considers prevention and mitigation strategies beyond design basis accidents, as well as common-mode failure, extreme external events, and severe accidents. It is able to prevent damage to the reactor core and to sustain containment integrity for 6 h after a potential accident without operator action. Human engineering requirement factors are sufficiently considered for reducing human effects. Combined with the above designs, CDF and LRF are less than 1.0×10^{-8} ry^{-1} (per reactor year), which complies with the safety requirements of GIII pressurized water reactors.

The engineered safety features of AES2006 are classified into four categories: protective, localizing, supporting, and control safety systems. Protective safety systems include high-pressure emergency injection, low-pressure emergency injection, primary circuit pressurizing, secondary circuit pressurizing, emergency gas removal emergency boron injection, and residual heat removal systems. Localizing safety systems are intended to prevent and limit the release and diffusion of radioactive substances, including a containment spray system. Supporting systems include emergency power supplies, safety systems process water supply, and MCR and SCR life support systems. Control safety systems are those intended for initiating safety system operation, their instrumentation and control during fulfillment of given functions. For a loss of coolant accident (LOCA), when the pressure is lower than 5.9 MPa, the passive ECCS accumulators are actuated before low-pressure emergency injection. A pilot-operated pressure relief valve mounted on the pressurizer prevents high-pressure core melt by discharging steam into the bubbler. For going beyond design basis accidents, if the active residual heat removal system cannot perform the design function, as a backup, AES2006 adopts a passive design for long-term residual heat removal through steam generators. Additionally, for beyond design basis accidents including severe core damage accidents, passive systems are adopted to reduce the containment pressure and temperature through long-term heat removal. Passive hydrogen recombiners are intended to control the concentration of hydrogen to prevent a hydrogen explosion. The reactor has a double containment for preventing radioactive substance release and resisting external events such as large commercial airplane crashes.

2.2.4 Economy

To decrease the initial investment of new plants, the following measures are taken: decreasing design man-hours, reducing the scope where verification is in need, shortening the construction duration, and maximizing the use of facilities, personnel, and bulk materials from the NPP construction country.

In addition, the following measures are taken to reduce operation costs: increasing equipment reliability and decreasing the amount of service maintenance, management costs of radioactive waste and spent fuel, water and electricity consumption during plant operation, production of chemical and other industrial waste, and so on. Moreover, strategies such as the improvement of plant output and thermal efficiency and the optimization of refueling are useful for improving the economic benefit of power plants.

2.3 AP1000 and CAP1400

2.3.1 Overview

Passive features rely on natural forces, such as gravity, natural circulation, and compressed air, to perform intended safety functions, including the emergency coolant injection, the decay heat removal, and the integrity of containment. Currently, to the extent possible, passive engineering safety features are adopted to deal with design basis accident or to prevent and mitigate the severe accident (Juhn *et al.*, 2000; Tujikura *et al.*, 2000; Zang *et al.*, 2001; IAEA, 2009; Krepper and Beyer, 2010; Zhang *et al.*, 2012).

AP1000 is a two-loop revolutionary GIII passive PWR developed by Westinghouse (Westinghouse, 2014). Compared with the traditional NPP, AP1000 adopts the passive safety concept. Revolutionary changes have taken place in the design of nuclear power plant safety systems due to the introduction of the passive concept. The design adopts a passive severe accident prevention and mitigation strategy. In the use of passive safety systems, operator action is not needed for 72 h after a postulated accident.

After the introduction of AP1000 technology, to further improve economic competitiveness, State Nuclear Power Technology Corporation (SNPTC) of China has developed CAP1400 with independent intellectual property rights. CAP1400 inherits the passive and simplified design concept of AP1000 NPP, and further improves electrical output. Figs. 4 and 5 show a schematic diagram of the reactor coolant system and the general layout of CAP1400, respectively (SNERDI, 2013). Compared with EPR and AES2006, the reactor coolant system is significantly simplified, and the layout is more compact. The preliminary design of CAP1400 has passed the review of National Energy Administration of China, and the preliminary safety analysis report (PSAR) has already passed the review of the National Nuclear Safety Administration of China.

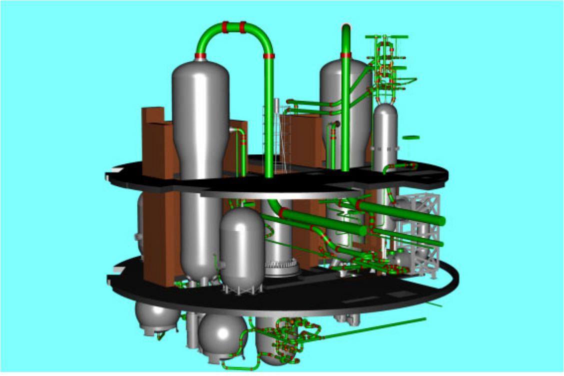

Fig. 4 CAP1400 reactor coolant system

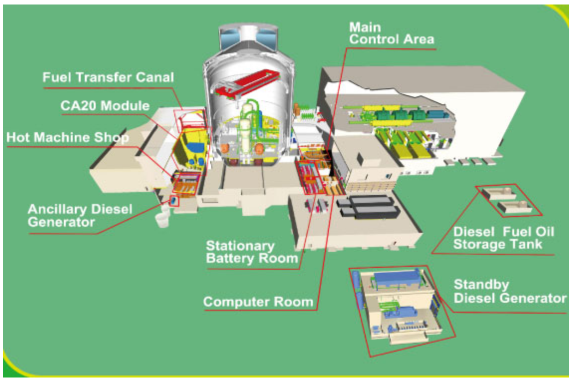

Fig. 5 General layout of CAP1400

2.3.2 Overall performance

The design life of CAP1400 is 60 years. Compared with AP1000, the thermal output has been increased from 3415 MW to 4058 MW, and the gross electrical output has been increased to 1500 MW. The steam pressure of the steam generator outlet has been increased from 5.61 MPa to 6.01 MPa. The availability is about 93%. The number of fuel assemblies has increased from 157 to 193. The fuel cycle length is 18 months, and it can be extended to 24 months. The reactor core is capable of loading MOX fuel. Mechanical shim (MSHIM) is used for load following to reduce the production of borated water. Ion exchanger and chemical coagulation treatment packages are used to minimize radioactive substances entering water in the surrounding environment, which satisfies the requirements of Chinese standards. It is capable of load following operation. The duration of construction from the first concrete date to connection to the grid is less than 56 months, made possible by the use of modular construction. The safe shutdown earthquake is 0.30*g*, which indicates extensive site applicability.

2.3.3 Safety

Similar to AP1000, CAP1400 adopts passive systems to withstand design basis accidents and to prevent and mitigate severe accidents, which enhances defense in depth. Passive safety systems perform functions without operator action, and they depend on gravity, natural circulation and compressed air, energy storage of batteries, and compressed fluid, rather than pumps, fans, emergency diesel generators, or other mechanical equipment. So safety-class support systems (such as alternating current (AC) power, component cooling water, service water, or heating, ventilation, and air-conditioning (HVAC)) are not used. In traditional active PWR, safety-class support systems are required. Actuations of passive engineered safety features need only the one-off action of a few valves. When the valves lose power supply or receive a safety actuation signal, they automatically open depending on direct current (DC) power supply or "fail-safe" design. CAP1400 has strategies for severe accident prevention and mitigation, and has established severe accident management guidelines to protect humans, society, and the environment from the harmful influence of radiation.

A passive core cooling system (Fig. 6) and passive containment cooling system (Fig. 7) are designed to withstand design basis accidents, by assuming no operator action within 72 h after accidents (SNERDI, 2013). For severe accidents, a passive containment cooling system, in-vessel retention (IVR, Fig. 8) (Rempe *et al.*, 2002; Knudson *et al.*, 2004), and a

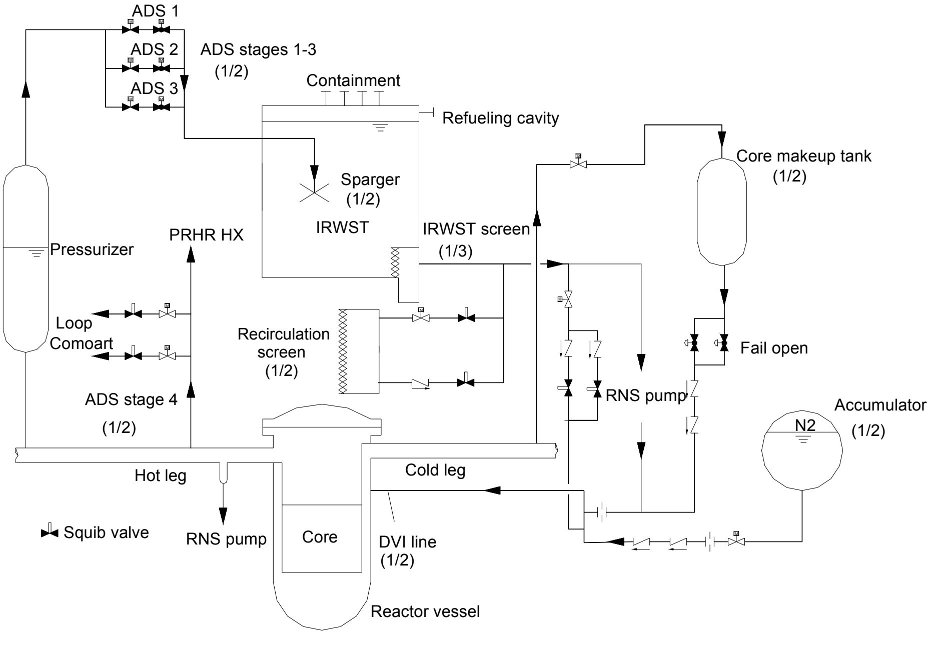

Fig. 6 Passive core cooling system

ADS: automatic depressurization system; IRWST: in-containment refueling water storage tank; PRHR HX: passive residual heat removal heat exchanger; DVI: direct vessel injection; RNS: normal residual heat removal system containment hydrogen control system (including passive hydrogen recombiner and igniters) have been designed. A passive main control room emergency habitability system is designed to provide a suitable environment for plant personnel in main control room. The application of human factors' engineering is enhanced and the human machine interface is optimized to minimize human error. A digital instrumentation and control system has been adopted to improve the accuracy of control and protection. The CDF is 4.00×10^{-7} ry^{-1} and the LRF is 5.07×10^{-8} ry^{-1}. A long-term water source and a portable power supply have been added into the CAP1400 design to improve the capability of passive PWRs to withstand extreme disasters, by learning from the experience of the Japanese Fukushima nuclear accident. After 72 h, the water in a passive containment cooling auxiliary water storage tank (PCCAWST) is supplied to a passive containment cooling water storage tank (PCCWST) by a recirculation or movable pump for 4 d of cooldown, so that containment integrity is maintained.

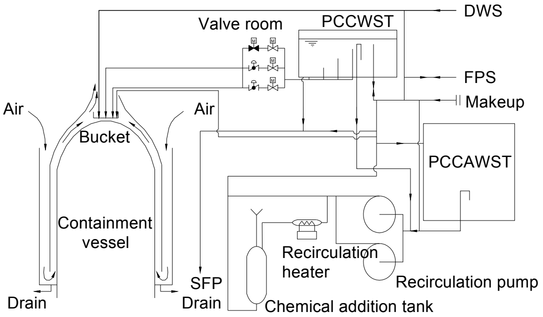

Fig. 7 Passive containment cooling system
DWS: demineralized water transfer and storage system; FPS: fire protection system; SFP: spent fuel pool

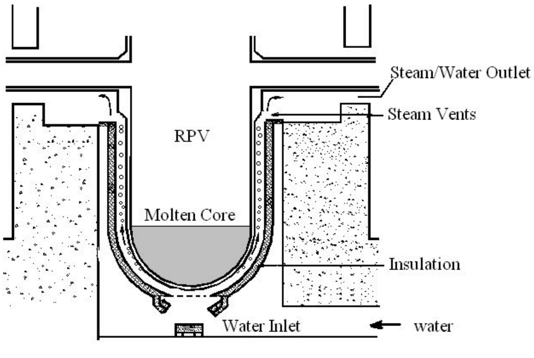

Fig. 8 In-vessel retention

2.3.4 Economy

Because the passive safety concept is adopted in the design of CAP1400, the safety system structures are simplified. The design reduces the number of safety support systems, significantly decreases the number of safety-class equipments and seismic buildings, and decreases the maintenance requirements. In addition, the 1E emergency diesel generator system and the majority of the safety-class active equipment have been removed, and the demand for bulk materials clearly decreases. Thus, CAP1400 has the characteristics of simplified design, simplified system structure, simplified layout, reduced workload, shortened construction duration, and decreased plant operation maintenance costs. The safety performance of CAP1400 is significantly improved, and CAP1400 has a strong competitiveness by reducing initial investment, shortening construction duration, decreasing spare parts, decreasing in-service inspections and tests, and improving availability of the plant. The economic benefit is enhanced by improving electrical output, availability, design life, and burnup. The duration of construction of CAP1400 will be shortened by incorporating feedback from AP1000 under construction in China and USA.

3 Discussion and analysis

It can be seen from Table 1 that, for the three typical GIII PWRs, design life has been prolonged from 40 to 60 years, availability has been improved by operation feedback and the reliability design, and unit power output has been improved to enhance economic competitiveness. The duration of fleet construction of CAP1400 has been reduced to 48 months, as the designers have learnt by feedback from units already constructed, which is less than that of EPR and AES2006 due to modular construction. Owing to the increase in installed capacity and the requirements of the grid, load following capability has been considered for the three PWRs, which will strengthen the flexibility of operation. The increase in the average discharge burnup of fuel and the length taken to refuel enhance fuel economy and plant availability. Additionally, the MOX fuel loading capability is in agreement with non-proliferation

Table 1 Typical performance comparison of EPR, AES2006, and CAP1400

Item	Description		
	EPR (Areva, 2014)	AES2006 (Rosatom, 2014)	CAP1400 (SNERDI, 2013)
Design life (year)	60	60	60
Availability (%)	>92%	>90%	>93%
Construction duration (from FCD to connection to grid) (month)	<54	<54	<56 (to be less for fleet construction)
Mode of operation	Baseload and load follow	Baseload and load follow	Baseload and load follow
Reactor thermal output (MW)	4590	3200	4040
Gross electrical output (MWe)	1770	1197	1500
No. of fuel assemblies	241	163	197
Average discharge burnup of fuel (MWd/tU)	>50 000	>60 000	>53 000
Fuel cycle length (month)	24	12, capable of 18–24-month refuel cycle	18, capable of 24-month refuel cycle
MOX fuel loading capability	Available	Available	Available
Average linear power density (kW/m)	16.67	16.78	18.10
No. of RCS loops	4	4	2
Average temperature of reactor (°C)	312.6	313.6	304.0
Steam generator type	U type, vertical	Horizontal	U type, vertical
Steam pressure of steam generator exit (MPa abs)	7.72	7.00	6.01
Flow rate of steam (kg/s)	2630	1780	2245
Feed water temperature (°C)	230	227	226.7
Engineering safety feature	Active	Active, passive	Passive
Operation action time (h)	<0.5	<6	<72
Severe accident mitigation	Dedicated severe accident depressurization valves, core catcher, containment spray system, passive hydrogen recombiner	Pilot-operated relief valves, passive heat removal system via steam generator, passive heat removal system from containment, core catcher, passive hydrogen recombiner	Passive depressurization squib valve, passive containment cooling system, in-vessel retention, passive hydrogen recombiner
Containment	Double-envelope: external, reinforced concrete; internal, pre-stressed concrete	Double-envelope: external, ordinary concrete; internal, pre-stressed reinforced concrete	Double-envelope: external, reinforced concrete; internal, steel
Safe shutdown earthquake	0.25g	0.25g	0.30g
Capability of withstanding large commercial aircraft crashes	Available	Available	Available
Core damage frequency, CDF (ry^{-1})	$<1.24\times10^{-6}$	$<5.94\times10^{-7}$	$<4.00\times10^{-7}$
Large release frequency, LRF (ry^{-1})	$<9.60\times10^{-8}$	$<3.70\times10^{-9}$	$<5.07\times10^{-8}$

FCD: the first concrete date; RCS: reactor coolant system requirements. The thermal efficiency of the three units is over 37% (gross); among which, EPR shows the best performance, due to the highest average reactor temperature, the adoption of an axial economizer for the steam generator to increase steam pressure, and optimization of the turbine.

The strategies for enhancing defense in depth are different for the three PWRs. For design basis accidents, both EPR and AES2006 enhance the reliability of their active engineering safety features to implement their intended function for potential accidents by following the principles of redundancy and diversity, which is realized by $N+3$ for active engineering safety features. For high-pressure and low-pressure injection, a pump is used to drive water into the reactor. On the other hand, a passive accumulator is adopted to implement the function of injection when the pressure of the primary loop is lower than a specified value. Decay heat removal is implemented by a safety-class normal residual heat removal system, a component cooling water system, and a service water system. Different from EPR and AES2006, CAP1400 adopts systematic passive engineering safety features, including coolant injection, decay heat removal, and depressurization, in order to deal with design basis accidents, which simplifies the system configuration. For small loss of coolant accidents (S-LOCA), two phases exist for decay heat removal. Before the actuation of ADS4, decay heat is mainly removed by feed-bleed; PRHR also takes some of the decay heat away. After the actuation of ADS4, containment sump long-term recirculation will be established and decay heat is removed by evaporation in the containment sump. From the perspective of defense in depth, before the actuation of ADS4, if the normal residual heat removal system is available, it can be used by following the specified procedure to remove decay heat to avoid the actuation of ADS4, which would lead to the large release of coolant into the containment. That is to say, CAP1400 can deal with accidents by level 2 of defense in depth; EPR and AES2006 would need to use level 3 of defense in depth to deal with S-LOCA.

For CAP1400, passive engineering safety features are adopted. However, it neither abandons nor relies on the engineering safety features and components of GII PWR for dealing with design basis accidents, such as diesel generators, normal residual heat removal systems, component cooling water systems, service water systems, or auxiliary feed water systems. These safety-important but non-safety-class systems and components are categorized as regulatory treatment of non-safety-related systems (RTNSS) which belong to level 2 of defense in depth. For the RTNSS systems and components, supplementary management requirements are added. Therefore, in the event of an accident, only if the RTNSS system or component failed would the accident progress.

As shown in Table 1, the requirements for operation action times for the three PWRs after postulated accidents are different. According to operation feedback, human-induced events are one of the sources which may lead to accidents. For accident conditions the operators are under high stress, so if they have enough time to understand what has happened, it will be helpful for them to make the correct decision. CAP1400 is able to prevent the reactor core from being damaged and sustain the integrity of containment for 72 h after initiation of an accident without operator action. This provides enough time for the operator to make correct judgments and thereby decrease human error; this is superior to the GIII PWR, which adopts active engineering safety features.

The three PWRs have systematic strategies for the prevention and mitigation of severe accidents, which is in accordance with the requirements of nuclear safety following the Fukushima nuclear accident. For the prevention of high-pressure core melt situations, the three PWRs have dedicated relief valves installed to eliminate the risk of high-pressure failure of the RPV, which would lead to direct containment heating with the potential for early containment failure. Auto-catalytic hydrogen recombiners are used for the three PWRs for hydrogen concentration control. Different from EPR and AES2006, CAP1400 adopts an IVR strategy to maintain the integrity of RPV by cooling the external surface of the vessel. EPR and AES2006 use a core catcher to confine melt debris within containment to protect the public and the environment from harmful radiation, which allows reactor core debris to penetrate the reactor pressure vessel and avoids MCCI.

For beyond design basis accidents, if the active decay heat removal system and containment cooling system are not able to perform their intended functions, as a backup, AES2006 utilizes a passive heat removal system via a steam generator and a passive heat removal system from containment to decrease the pressure and temperature within containment to keep it intact, which is different from EPR. From Table 1, it can be seen that the LRF of AES2006 is less than those of EPR and CAP1400, which demonstrates that the active plus passive strategy for severe accident mitigation is effective.

Due to the characteristics of passive philosophy, passive systems need limited components and have less dependence on the environment when they perform their intended functions, especially given higher reliability for mitigation of severe accidents induced by extreme external events. That is one of the reasons why EPR, AES2006, and CAP1400 adopt the passive philosophy to different degrees to deal with design basis accidents or to prevent and mitigate severe accidents.

All the three PWRs adopt double-envelope containment to confine radioactive substances and to protect the plant from external hazards. CAP1400 establishes a new "cooling chain" with the internal containment being made of steel, which is different from the traditional method for removing decay heat. Due to CAP1400's cooling chain, the traditional normal residual heat removal system, the component cooling water system, and the service water system are not safety related; these are still safety-class for EPR and AES2006. Compared with GII PWR, the seismic capability of the three GIII PWRs has been enhanced. Superior to EPR and AES2006, the safe shutdown earthquake of CAP1400 is 0.30*g*, which strengthens the safety of the plant if an earthquake occurs and broadens the potential site. Furthermore, due to the influence of the "911" disaster, the three PWRs are capable of resisting large commercial airplane crashes.

4 Conclusions

From the comparison above, it can be found that the typical GIII pressurized water reactors, EPR, AES2006, and CAP1400, focus on improving overall performance, safety levels, and economic competitiveness to obtain support from the government, public, and investors. Detailed improvements cover: prolonging design life by improving component performance, increasing plant availability through reliability design, increasing the reactor output and improving thermal efficiency, increasing fuel economy by improving fuel performance, shortening the duration of construction by adopting advanced construction methods, and satisfying the load following requirements. In the future, the main trends in GIII pressurized water reactor technology may be in the following fields:

1. It is necessary to strengthen the defense in depth strategy to improve safety levels and cover the design of extended station coping strategies, such as in the case of blackout (SBO), airplane crashes, and anticipated transients without scram (ATWS). Moreover, extreme external events, including earthquakes and flooding, should be taken into consideration (ASME, 2012; IAEA, 2012).

2. It would be meaningful to deepen studies of the pros and cons of passive and active safety approaches by improving the understanding of nuclear power plant safety from top-tier design philosophy and to scan and analyze the factors which may systematically impact their safety (Zio *et al.*, 2010). Based on the work mentioned, it is necessary to combine passive and active features to improve the safety performance of nuclear power plants; it is important to understand how to make use of passive and active approaches to deal with design basis accidents, and to prevent and mitigate severe accidents.

3. After the Fukushima accident, the National Nuclear Safety Administration of China issued the "Twelfth Five-Year Plan" and the "2020 Vision of Nuclear Safety and Radioactive Pollution Prevention and Mitigation" (MEP, 2012); it is mentioned that: "for siting and design, the latest nuclear safety codes and standards issued by Chinese government and IAEA should be followed, the reactor technology which is more matured and advanced should be adopted for construction", "the advanced reactor technology with our own intelligent proprietary right and high safety performance should be developed actively." It should be strived to eliminate the possibility of large release of radioactive substance practically from design for China's "Thirteenth Five-Year Plan" and new-build nuclear power plant. It can be found that for GIII nuclear power plants, especially pressurized water reactors, adopting passive safety features will be the main trend for new-build nuclear power plants. Furthermore, it is urgent to work out the technological essence of the phrase "to eliminate the possibility of large release of radioactive substance

practically", including an accurate definition of "eliminate practically" and a quantitative index for "large release", from which it will be possible to form the specified technology by systematically enhancing the defense in depth of nuclear power plants.

4. The applicability of components and instruments for severe accident mitigation and management has become a great concern since the Fukushima accident. It would be very helpful to scan the components and instruments needed for severe accident mitigation by following the severe accident mitigation strategy and to demonstrate that these components and instruments could implement their intended functions in severe accident circumstances.

5. It is important to improve the economic competitiveness of GIII PWRs and make sure construction costs will not exceed budget by enhancing design simplification and standardization, incorporating the first of a kind (FOAK) construction experiences of EPR, AES2006, and CAP1400 into the *N*th of a kind (NOAK) unit construction, and optimizing constructability and modularization, which would determine the market of EPR, AES2006, and CAP1400 in the future.

6. From the perspective of operation, attention should be paid to the load following capability of nuclear power plants due to increasing domestic installed capacity, long design life, and the influence of renewable energy when connecting to the grid.

7. In the long term, the impact of possible climate change should be taken into consideration for nuclear power plants to make sure that they are capable of dealing with extreme weather (OECD, 2014), such as high environment temperatures and increased cooling water temperatures.

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中文概要

题 目：世界三代压水堆主要机型技术分析

概 要：本文针对目前主要的第三代压水堆机型 EPR、AES2006 以及 CAP1400，从核电厂设计理念，综合性能、安全性和经济性等方面进行对比分析。在此基础上，结合福岛核事故，探讨中国商用核电未来的技术发展方向。通过对 EPR、AES2006 以及 CAP1400 的对比分析发现，尽管它们采用了不同的设计理念和技术方式，但作为目前第三代商用压水堆的代表机型，其目标都在于提高核电厂的整体性能、安全性能和经济性，从而在提高安全性的前提下，强化其相比于其他发电方式的竞争力，获得政府、公众和业主的支持。第三代压水堆技术的主要努力方向在于：在安全方面，进一步强化纵深防御体系，将设计加强工况（包括全厂断电、商用飞机撞击和预期未能停堆的瞬态等）纳入设计考虑的范畴；设置预防和缓解严重事故的措施；考虑极端外部事件设防（包括地震和洪水等）。再者，对非能动安全与能动安全两者之间的关系定位、相互衔接进行优化设置，从而更好的保障核安全。此外，严重事故下设备和仪表的可用性成为福岛核事故后需要特别关注的问题。从经济性的角度讲，加强设计简化和标准化，及时将 EPR、AES2006 以及 CAP1400 的首台组的建造经验反馈到后续机组，改进可建造性和模块化从而确保经济性和缩短建造周期。从运行的角度讲，考虑国内核电装机容量的增加、较长的设计寿命以及其他可再生能源的并网，核电设计需强化负荷跟踪的能力。长期来看，需要考虑可能的气候变化，从而确保核电站（沿海和内陆）具有较强的应对极端气候以及较高环境温度和冷却水升温的能力。

关键词：第三代压水堆技术；性能；安全性；经济性；核电