Status report 98 - Advanced Boiling Water Reactor II (ABWR-II)

Overview

Full name	Advanced Boiling Water Reactor II
Acronym	ABWR-II
Reactor type	Boiling Water reactor (BWR)
Coolant	Light Water
Moderator	Light water
Neutron spectrum	Thermal Neutrons
Thermal capacity	4960.00 MWth
Gross Electrical capacity	1717.00 MWe
Design status	Basic Design
Designers	GE-Hitachi
Last update	21-07-2011

Description

Introduction

By adopting a large electric output (1700 MWe), a large fuel bundle, a modified Emergency Core Cooling System, and passive heat removal systems, and other design features, a design capable of increasing both economic competitiveness and safety performance is achieved. The key objectives of ABWR-II are further improvement in economics against alternative forms of generation and enhancement of safety & reliability. The design goals are:

- Economic competitiveness
  - Power generation cost: 20% cost reduction from standardized ABWR
  - Overnight capital cost: 30% cost reduction from standardized ABWR
  - Construction period: 29.5 months (from first concrete work to fuel loading)
- Safety and reliability
  - Good combination of active and passive systems
  - Provision of grace period both for transients and accidents: one day grace period
  - Consideration of severe accident from design stage
  - Refinement of PSA performance (equal to or higher than that of ABWR, especially on containment

capability)

- Sustainability for future fuel cycle uncertainty
  - Increased flexibility for higher burn-up, MOX and higher conversion

2.1 Primary circuit and its main characteristics

The primary functions of the ABWR-II nuclear boiler system are:

1. To deliver steam from the reactor pressure vessel (RPV) to the turbine main steam system;
2. To deliver feedwater from the condensate and feedwater system to the RPV;
3. To provide overpressure protection of the reactor coolant pressure boundary; and,
4. To provide automatic depressurization of the RPV in the event of a loss of coolant accident (LOCA) where the RPV does not depressurize rapidly.

Main steam lines (MSLs) are designed to direct steam from the RPV to the main steam system of the turbine, and feedwater lines to direct feedwater from condensate and feedwater system to the RPV.

The main steam line flow limiter, a flow restricting venturi built in the RPV MSL nozzle of each of the four main steam lines, limits the coolant blowdown rate from the reactor vessel to a choked flow rate equal to or less than 200% of rated steam flow at 7.07 MPa upstream gauge pressure in the event when a main steam-line break occurs anywhere downstream of the nozzle.

There are two main steam isolation valves (MSIVs) welded into each of the four MSLs: one MSIV inside the containment and one MSIV outside the containment. The MSIVs are Y-pattern globe valves. The Y-pattern configuration permits the inlet and outlet flow passages to be streamlined to minimize pressure drop during normal steam flow.

The new type MSIV for ABWR-II, now under development, is illustrated in Figure 2.1. The bore diameter is increased and the center of gravity of its driving mechanism is lowered. This bore diameter increase is not simply an enlargement from ABWR but is optimized so that the pressure loss is decreased to increase plant efficiency. The lowered center of gravity of the driving mechanism by relocating springs and an oil damper will contribute to improvement in seismic capability.

The nuclear pressure relief system consists of safety/relief valves (SRVs) located on the MSLs between the RPV and the inboard MSIV. There are 14 SRVs distributed on the four MSLs. The SRVs are designed to provide three main protection functions: overpressure safety; overpressure relief; and, depressurization operation.

For ABWR-II, an increase of discharge capacity and simplification of valve structure were considered. To decrease the number of SRVs, the discharge capacity per SRV is increased by 70 % to 680 t/h (nominal) from ABWR's 395 t/h with an increased throat diameter and increased coil spring diameter. Also, the structure of the SRV is simplified by integrating an air cylinder into the SRV main body, as illustrated in Figure 2.2. The developmental test program for the new SRV is proceeding.

The automatic depressurization subsystem (ADS) consists of the six SRVs and their associated instrumentation and controls. The ADS designated valves open automatically for events involved with small breaks in the nuclear system process barrier or manually in the power actuated mode when required. The ADS designated valves are capable of operating from either ADS LOCA logic or overpressure relief logic signals. The ADS accumulator capacity is designed to open the SRV against the design drywell pressure following failure of the pneumatic supply to the accumulator.

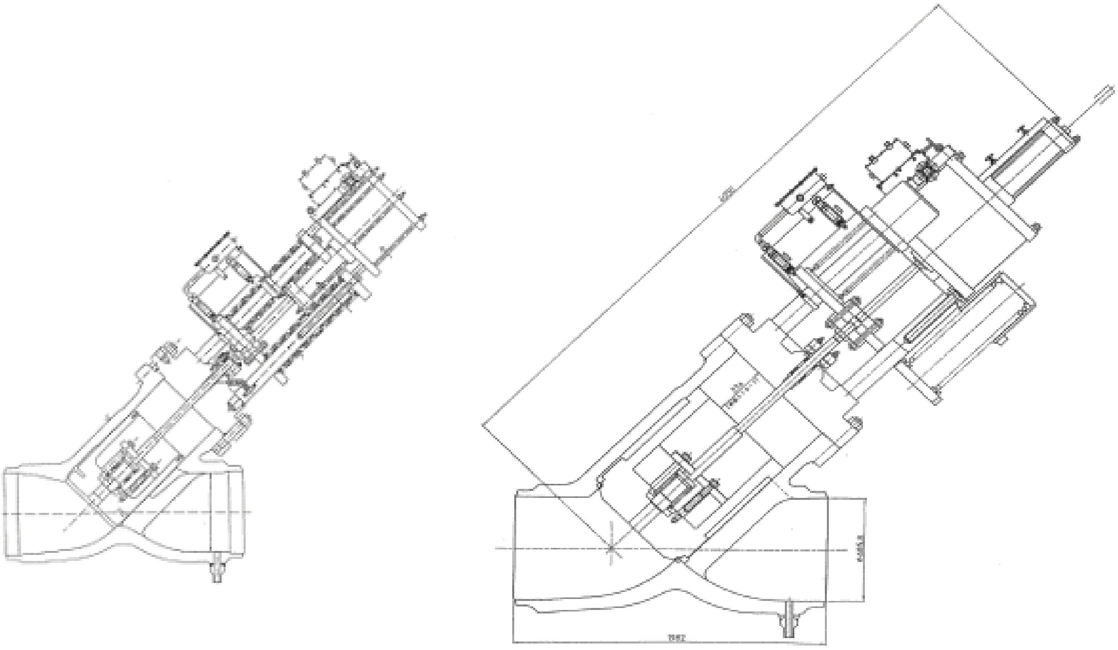


FIG.2.1. MSIVs for ABWR (left) and ABWR-II (right).

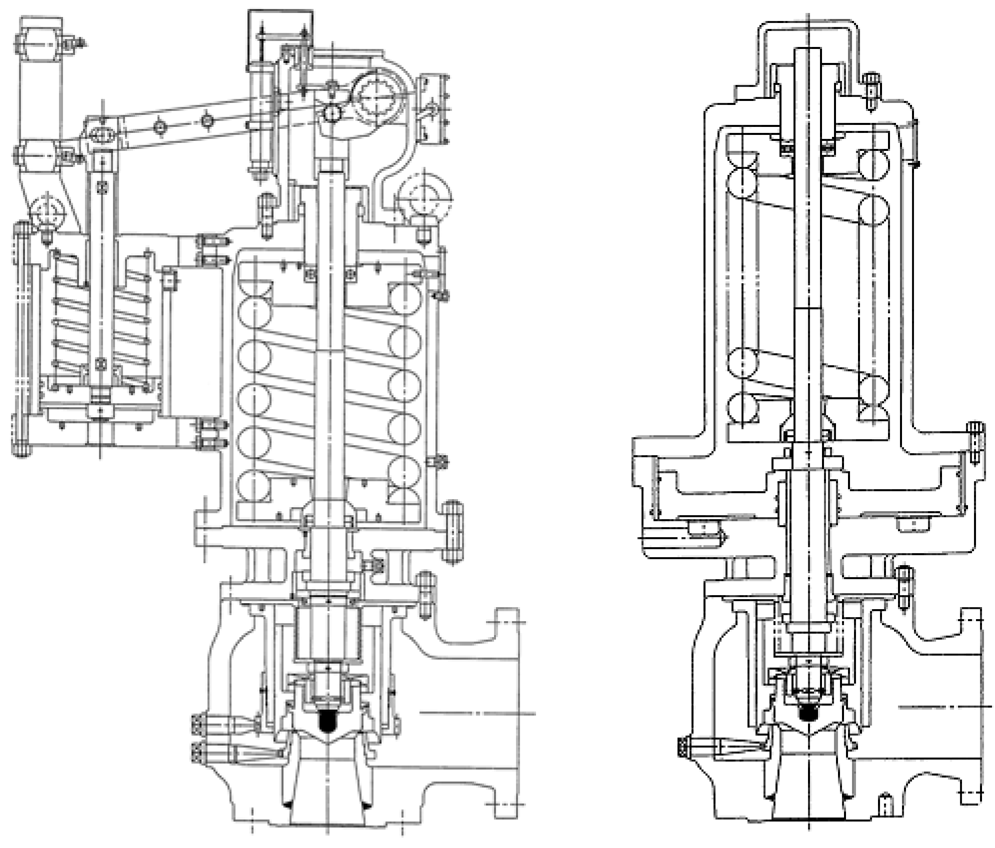


FIG. 2.2. SRVs for ABWR (left) and ABWR-II (right).

2.2 Reactor core and fuel design

Reference design

Basic policies of the ABWR-II core design are as follows:

- Keep the same level margin as the current core design in order to have enough flexibility for future higher burnup and longer cycle operation, under the condition to meet power uprate to 1700 MWe;
- Reduce components and shorten the refueling outage time to improve the capacity factor.

To meet above requirements, 1.5 times larger fuel bundle and K-lattice were selected for the reference core design. The large bundle is able to increase the area inside the channel box and has potential for increasing the number of fuel rods in support of uprating plant output. On the other hand, the fuel bundle pitch increase results in less cold shutdown margin (CSDM) due to decrease of the number of CRs. The K-lattice control rod concept was chosen as a countermeasure for cold shutdown margin. This concept, compared to the conventional control rod design (N-lattice), is illustrated in Figure 2.3. In the K-lattice concept, the number of CRs per bundle is increased to two CRs for every four fuel bundles, while there is one CR per four bundles for the conventional lattice. The K-lattice control concept provides improvement in CSDM and makes it possible to adopt a larger bundle.

The specifications of the ABWR-II core are listed in Table I and the configuration of the core is displayed in Figure 2.4. The thermal output of the ABWR-II core is 4960 MW, 1.26 times larger than that of the ABWR core. Although the former has 424 bundles, which is 49% of that of the ABWR, there are 197 control rods (CRs), about the same as in the ABWR. This is because the ABWR-II employs the K-lattice in order to maintain the cold shutdown margin (CSDM). However, the ABWR-II has fewer CRs per unit output than the ABWR.

Targeted operating cycle length and average discharge burnup are set to be 18 EFPM and 60 GWd/t, respectively. The recirculation flow control range of 20% is achieved.

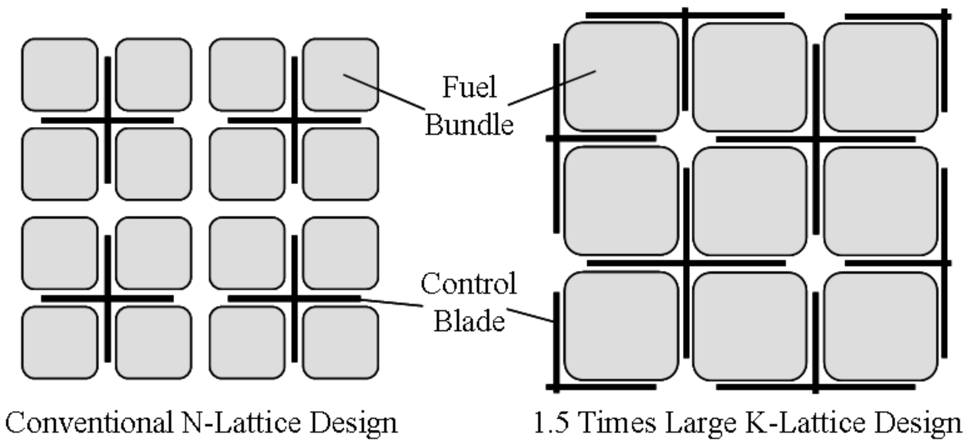


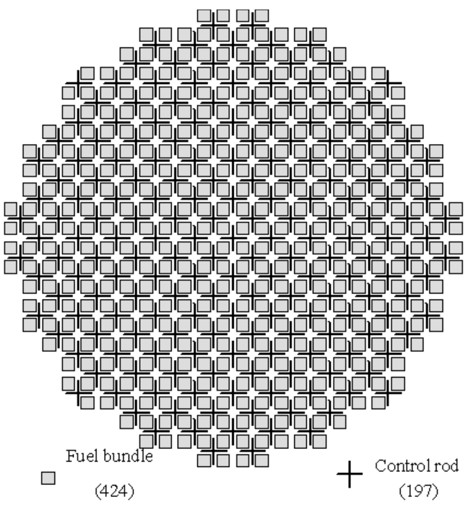
FIG 2.3. Comparison of Lattice Configurations

Item	Un	ABW	AB
Electric	M	170	135
Reactor	M	496	392
Operating	EFP	1	1
Average	GW	6	4
Maximum core	t/h	62.1x^{3}	57.9x^{3}
Active core	m	3.7	3.7
Fuel bundle	c	23.	15.
Number of	-	19	20
Number of fuel	-	42	87

TABLE 2.1. MAIN PARAMETERS FOR ABWR-II CORE

The main characteristic of the ABWR-II bundle is 1.5 times larger bundle pitch compared to the conventional BWR bundle. This large bundle comprises four sub-bundles, each of which consists of an 8 x 8 fuel arrangement.

In the reference design, shown in Figure 2.5, the four sub-bundles are separated by a partition and there is a large water box in the center, formed by parts of the partition, which occupies 24 fuel rod positions. In addition, there are eight small water rods, each equivalent in size to the fuel rod.



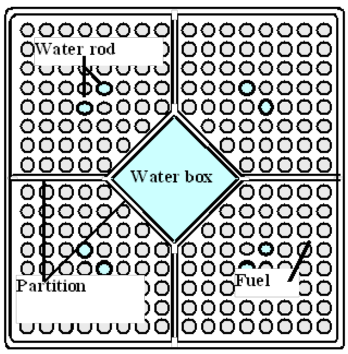


FIG. 2.4. ABWR-II core configuration configuration FIG 2.5. Reference

Spectral shift rod (SSR)

In the BWR, excess reactivity is controlled by burnable absorbers, CRs and recirculation flow. The recirculation flow can control the reactivity through the void fraction, which changes the neutron spectrum. In order to reduce use of CRs under ordinary operation and improve neutron economy, it is desirable to enlarge the controllable reactivity using recirculation flow. The reference core design has a more negative void coefficient due to larger fuel inventory and reactivity control capability with core flow becomes larger.

In order to enlarge the reactivity change due to unit recirculation flow, the SSR has been developed. The SSR is used instead of the water rod, in which the water level develops naturally during operation and changes according to recirculation flow through the channel. The SSR configuration, illustrated in Figure 2.6, consists of a large ascending path and a narrow descending path. The coolant enters the ascending path from below the lower tie plate (LTP), goes up to the top, comes down in the descending path, and goes out right above the LTP. With this structure and a tight inlet, the water flow rate into the SSR is so small that the water heated by the irradiation of gamma rays and neutrons reaches the saturation temperature and boils in the SSR. Since the water velocity is very small in the ascending path, a water level develops there. In this system, the pressure drop caused by the coolant passing through the LTP almost equals the main part of the static head of the water column in the ascending path, which is the part from the height of the exit hole to the height of the water level. The pressure drop at the LTP is nearly proportional to the second power of the flow rate through it. Thus, by changing the recirculation flow rate, the water level in the SSR can be varied substantially.

Water shifting of the SSR also affects the average void fraction. Earlier in the cycle, the water level in the SSR is low, hence the peak of axial power shape tends to be located at the bottom half, where more water is present. Usually the average void fraction earlier in the cycle is higher due to the low recirculation flow rate. Then this axial power shape enhances that. Then, later in the cycle as the water comes up in the SSR, the peak of the axial power moves to the upper half, which makes the average void fraction much lower. By a synergistic effect of the water level change in the SSRs and an enhanced void fraction change in the channels, the reactivity change by the recirculation flow rate can be enlarged. Fuel utilization is improved by enhanced spectral shift operation, namely a harder neutron spectrum at BOC and a softer neutron spectrum at EOC than ordinary BWRs.

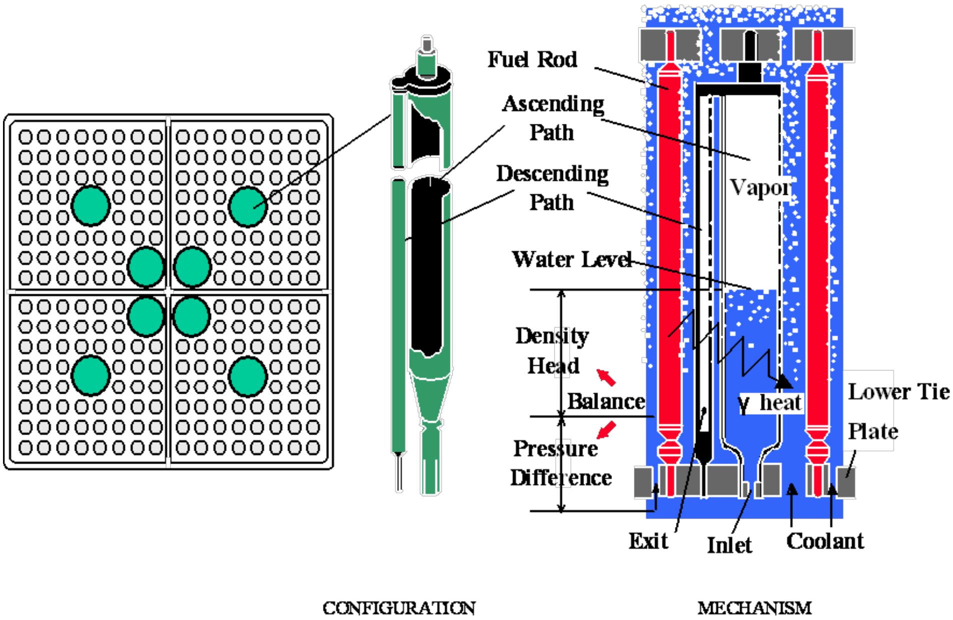


FIG. 2.6. Configuration and operational principle of SSR

In the case of the ABWR-II assembly, which has a larger in-channel space, a larger volume fraction of the SSR is affordable compared with the conventional BWR assembly. The results of the equilibrium cycle analyses with 18-month cycle operation and discharge burnup of 60 GWd/t showed that the SSR core design could allow operation with all CRs withdrawn throughout the entire cycle without increasing the maximum core flow rate. It was also shown that the uranium saving factor of about 6-7% against the reference ABWR-II core could be expected due to the higher spectral shift effect and lower Gd enrichment design. By incorporating the SSRs into the ABWR-II, the capacity factor can be increased because the necessity for CR and CR drive inspections is lessened. Moreover, the operation becomes much easier due to no need for a CR strategy and operation. The combination of these advantages with higher fuel utilization means the ABWR-II with SSRs should be an attractive alternative for the next generation nuclear reactor.

Control rod drive system

The control rod drive (CRD) system is composed of three major elements: the fine motion control rod drive (FMCRD) mechanisms; the hydraulic control unit (HCU) assemblies; and, the control rod drive hydraulic (CRDH) subsystems.

Compared with ABWR design, an improvement in the connection between the CRD motor and the CRD shaft was made. The improved mechanism is called a magnetic coupling. The magnetic coupling can transmit torque between the CRD motor and the CRD shaft through the pressure boundary without a penetration instead of having a seal around a penetrating shaft. This improvement eliminates the sealing parts where inspection and maintenance are most necessary, and also the seal detection system that requires monitoring during operation, thus making CRDs maintenance free, contributory to maintenance cost reduction.

2.3 Fuel handling and transfer systems

The reactor building is supplied with a refueling machine for fuel movement and servicing. The fuel handling and transfer system for ABWR-II is basically the same as on ABWRs except the fuel assembly weight of ABWR-II is heavier than that of ABWR.

2.4 Primary components

Reactor pressure vessel

The reactor pressure vessel (RPV) system consists of (1) the RPV and its appurtenances, supports and insulation, and (2) the reactor internal components enclosed by the vessel, excluding the core (fuel assemblies, control rods, in-core nuclear instrumentation and neutron sources), reactor internal pumps (RIPs), and control rod drives (CRDs). The RPV system is located in the primary containment. The reactor coolant pressure boundary portion of the RPV and its appurtenances act as a radioactive material barrier during plant operation.

Certain reactor internals support the core, flood the core during a loss of coolant accident (LOCA) and support safety related instrumentation. Other RPV internals direct coolant flow, separate steam, hold material surveillance specimens, and support instrumentation utilized for plant operation.

The RPV system provides guidance and support for the CRDs. It also distributes sodium pentaborate solution when injected from the standby liquid control (SLC) system.

The RPV system restrains the CRD to prevent ejection of the control rod connected with the CRD in the event of a failure of the reactor coolant pressure boundary associated with the CRD housing weld. A restraint system is also provided for each RIP in order to prevent the RIP from becoming a missile in the event of a failure of the reactor coolant pressure boundary associated with the RIP casing weld.

The RPV is a vertical, cylindrical vessel of welded construction with removable top head and head closure bolting seals. Through the use of large forged rings, the number of welds in the RPV is reduced. The main body of the installed RPV has a cylindrical shell, flange, bottom head, RIP casings, penetrations, brackets, nozzles, and the shroud support, which has a pump deck forming the partition between the RIP suction and discharge. The shroud support is an assembly consisting of a short vertical cylindrical shell, a horizontal annular pump deck plate and vertical support legs.

An integral reactor vessel support skirt supports the vessel on the reactor pressure vessel pedestal. Anchor bolts extend from the pedestal through the flange of the skirt. RPV stabilizers are provided in the upper portion of the RPV to resist horizontal loads. Lateral supports for the CRD housings and in-core housings are provided.

Reactor internals

The major reactor internal components in the RPV system are: (1) core support structure, and (2) other reactor internals.

The core support structures encompass: the shroud, shroud support and a portion of CRD housings inside the RPV, core plate, top guide, fuel support, and control rod guide tubes (CRGTs).

Other reactor internals are:

- Feedwater spargers, shutdown cooling (SDC) and low pressure core flooder (LPFL) spargers for the residual heat removal (RHR) system, high pressure core flooder (HPCF) spargers and couplings, and a portion of the in-core housings inside the RPV and in-core guide tubes (ICGTs) with stabilizers;

- Surveillance specimen holders, shroud head and steam separator assembly and the dryer assembly.

Reactor recirculation pumps

The reactor recirculation system features an arrangement of ten variable speed reactor coolant recirculation pumps. The pumps with motors are mounted in the bottom of the RPV, and are thus termed reactor internal pumps (RIPs). The RIPs provide forced circulation of the reactor coolant through the lower plenum of the reactor and up through the lower grid, the reactor core, steam separators, and back down the downcomer annulus.

The recirculation flow rate is variable over a "flow control range", from minimum flow established by certain pump performance characteristics to above maximum flow required to obtain rated reactor power.

Each RIP includes a device that prevents reverse RIP motor rotation by reverse flow induced torque. The RIP motor cooling is provided by an auxiliary impeller mounted on the bottom of the motor rotor, which circulates water through the RIP motor and its cooling heat exchanger. The heat exchangers are cooled by the reactor building closed cooling water (RCW) system.

2.5 Reactor auxiliary systems

The main auxiliary systems in the ABWR-II nuclear island consist of the residual heat removal (RHR) system, the reactor building closed cooling water (RCW) system, the reactor building seawater (RSW) system, the reactor water cleanup (CUW) system, the fuel pool cooling and cleanup (FPC) system and the suppression pool cleanup (SPCU) system. In addition there are many other auxiliary systems such as instrument and service air, condensate and demineralized water transfer, HVAC, equipment drain, floor drain and other systems which are basically the same as on ABWR plants.

Optimization was made in RHR system together with RCW and RSW systems. Taking into consideration that the passive heat removal systems of ABWR-II can be counted as a backup, the basic system configuration of RCW is two divisions instead of the three in ABWR. For RHR, RSW and active components in RCW in total make up four-division configuration that facilitates on-line maintenance and increases reliability and safety.

As to emergency power sources for active components in RHR/RCW/RSW systems, a four-division configuration consisting of two diesel generators and two gas turbine generators is applied to increase diversity and to facilitate maintenance. On-line maintenance will be applied to the diesel generators. The gas turbine generators are expected to be maintenance free.

The CUW system consists of piping, valves, pumps, heat exchangers and filter demineralizers that remove impurities from the reactor primary coolant water to maintain water quality within acceptable limits during the various plant operating modes. The CUW design for ABWR-II is basically the same as on ABWRs.

The FPC and SPCU systems consist of piping, valves, pumps, heat exchangers and filter demineralizers that are used to remove decay heat from the spent fuel storage pool and to remove impurities from the water in the spent fuel pool and dryer/separator pool and suppression pool to maintain water quality within acceptable limits during the various plant operating modes. The FPC and SPCU systems are basically the same as on ABWRs.

Description of safety concept

3.1 Safety requirements and design philosophy

The safety related requirements established during early phases of ABWR-II development are:

- Good combination of active and passive systems;
- Provision of grace period both for transients and accidents;
- Consideration of severe accident from design stage;
- Refinement of PSA performance (equal to or higher than that of ABWR, especially on containment capability).

Considering these requirements, ABWR-II design provides more emphasis on beyond-DBA capability in order to achieve a high level of safety such as the practical exclusion of the probability of emergency evacuation/resettlement. Optimization of safety and economic aspects is also to be strongly pursued. In order to accomplish these objectives, the following design approach was taken:

- Systems important to safety are incorporated in an integrated manner;
- Hardware increase is minimized for cost dominant portion;
- Additional benefits are introduced, as much as possible.

The safety related system configurations and their performance are described in the following sections.

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

The current reference concept includes the following safety related system design features:

- Rationalized four division RHR
- Diversified emergency power supply
- Reactor core isolation cooling (RCIC) system with a generator
- Passive heat removal systems

The ABWR-II ECCS configuration is shown in Figure 3.1. Cooling water injection system is comprised of the high-pressure core flooder (HPCF) and low-pressure flooder (LPFL). Optimization was made in RHR system together with reactor building closed cooling water (RCW) system and reactor building seawater (RSW) system. Taking into consideration that the passive heat removal systems of ABWR-II can be counted as a backup, the basic system configuration of RCW is two divisions instead of the three in ABWR. This two-division configuration is expected to reduce equipment cost for RCW that has relatively large amount of materials especially for piping. For RHR, RSW and active components in RCW in total make up four-division configuration that facilitates on-line maintenance and increases reliability and safety.

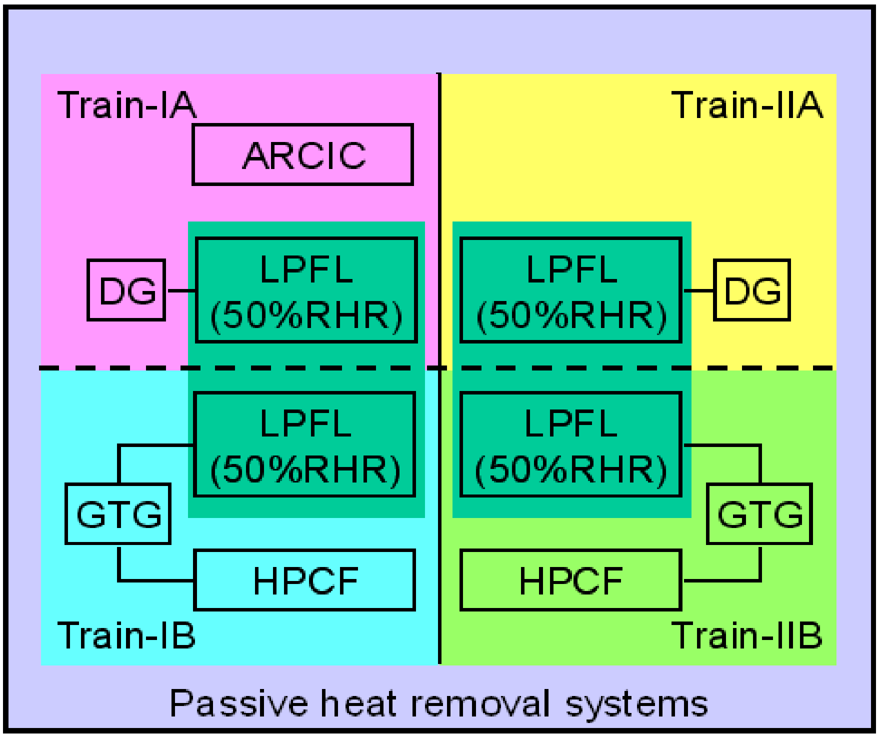


FIG. 3.1. ABWR-II ECCS configuration

As to emergency power sources for active components in RHR/RCW/RSW systems, a four-division configuration consisting of two diesel generators and two gas turbine generators is applied to increase diversity and to facilitate maintenance. On-line maintenance will be applied to the diesel generators. The gas turbine generators are expected to be maintenance free. Therefore, increased reliability and a reduced maintenance outage period will be achieved by this optimized division combination of two and four.

Since large break LOCA has been eliminated by adopting the RIP, LOCA is not the limiting event for ECCS capacity. Actually, high-pressure injection system capacity is determined from reactor water level set point requirements during transients such as loss of feedwater, and low-pressure injection system capacity is a result of optimum balance of residual heat removal system design. Utilizing these injection systems as ECCS, core coverage throughout the entire LOCA spectrum is achieved. Figure 3.2 shows an example of reactor water level transient during typical LOCA assuming not only single failure but also on-line maintenance for one train of the low-pressure injection system.

Containment design employs conventional pressure suppression as proven and cost-effective technology. Flow capacity of vent pipes and SRV discharge lines are increased from those of current ABWR, reflecting increased power, and the large capacity SRV quencher design resolves layout restriction in the suppression pool. Suppression pool water inventory is determined considering heat sink capacity requirements for all design basis events (LOCA blowdown and SRV discharge during reactor isolation event).

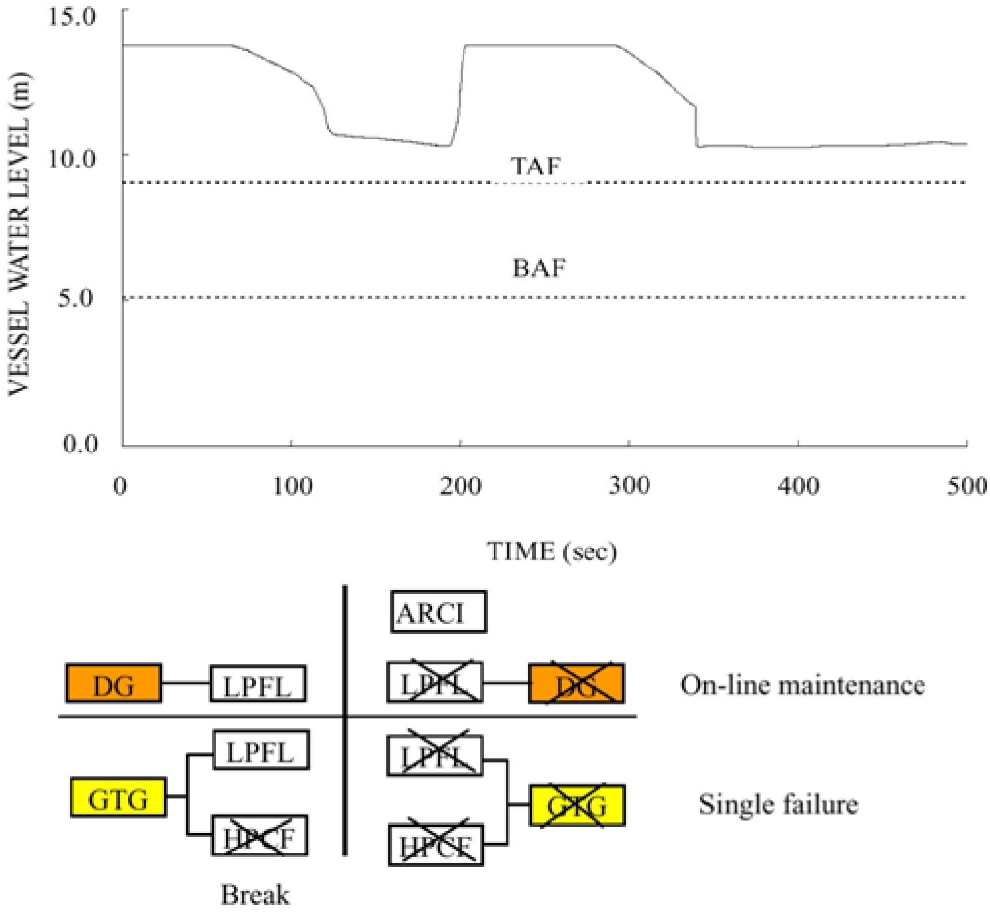


FIG. 3.2. The result of the DBA LOCA analysis by SAFER code

3.3 Severe accidents (beyond design basis accidents)

ABWR-II ECCS network has in-depth capability of redundant high-pressure indection similar to that of ABWR, with extended capability. The advanced reactor core isolation cooling (ARCIC) system has capability of self-standing operation and power supply under long-term station blackout (SBO) codition beyond battery capacity. In-depth inventory makeup is performed by HPCF as a backup of ARCIC for loss of feedwater event. In the event that emergency operating procedure is called, any single ECCS pump can maintain fuel cladding temperature and oxidation below PSA success criteria (1200 °C and 15 %) utilizing depressurization system as needed.

One of the new features of ABWR-II safety design is adoption of passive systems. The passive heat removal system (PHRS) consists of two dedicated systems, namely passive reactor cooling system (PRCS) and passive containment cooling system (PCCS), and common heat sink pool above the containment allowing one-day grace period (Figure 3.3). These passive systems not only cover beyond DBA condition, but also provide in-depth heat removal backup for RHR, and practically eliminate necessity of containment venting before and after core damage as a means of overpressure protection.

Flammable gas control in the containment is performed by the combination of inerting and passive autocatalytic recombiner (PAR), which has advantages both of safety (automatic startup and passive operation) and economy (low cost, flexible layout and easy maintenance).

The containment design considers severe accident phenomena such as direct containment heating (DCH), fuel coolant interaction (FCI), and molten core concrete interaction (MCCI) on a safety margin basis. The Japanese industry, collaborating with experts in research organizations, has recently established guidelines for containment performance design/evaluation under severe accident, and detailed quantitative examination from both phenomenological and probabilistic aspects is underway.

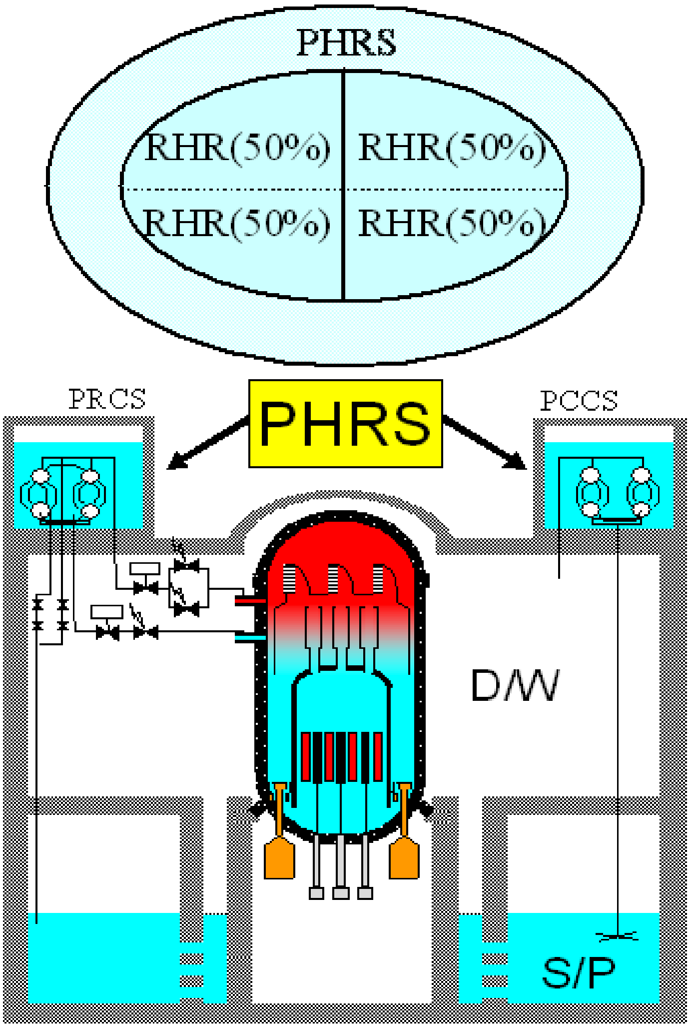
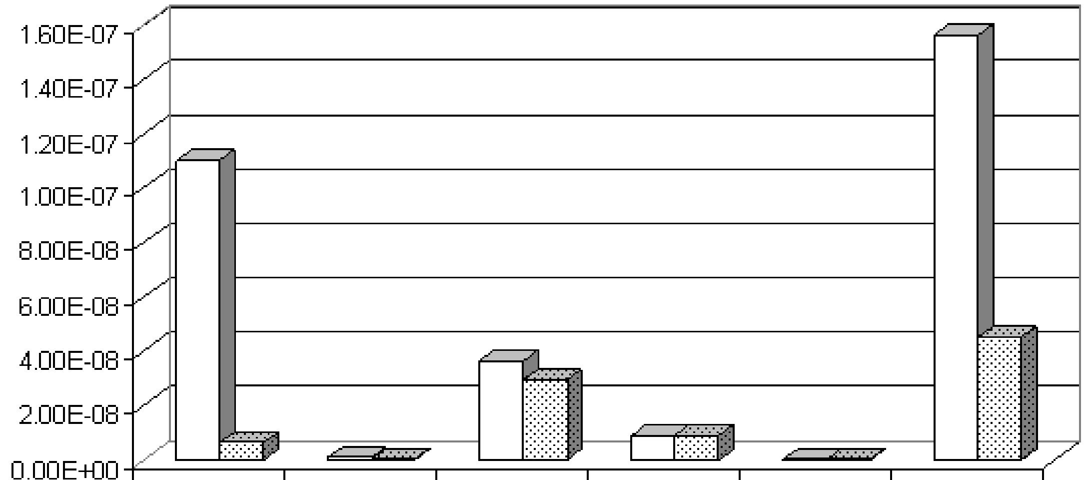


FIG. 3.3. Passive heat removal system



	SBO	LOCA	Transients	LOOP	ATWS	Total CDF
SSAR	1.10E-07	6.80E-10	3.66E-08	9.04E-09	2.70E-10	1.57E-07
ABWR II	6.19E-09	2.68E-10	2.94E-08	9.10E-09	2.19E-10	4.52E-08

FIG. 3.4. Results of level 1 PSA for the ABWR-II and the ABWR

Proliferation resistance

No information provided.

Safety and security (physical protection)

No information provided.

Description of turbine-generator systems

6.1 Turbine generator plant

The main turbine

The main turbine is six flow, tandem compound, reheat 1500 rpm machine with 1320.8mm (52 in.) last stage blades. The turbine has one dual-exhaust high-pressure section and three dual-exhaust low-pressure sections. The cycle uses moisture separators and reheaters with reheat for the cross-around steam.

Extraction steam from the high and low-pressure turbine extraction nozzles is conveyed to the high and low-pressure feedwater heaters. The feedwater heating systems are designed to provide a final feedwater temperature 216 °C (420°F) at 100 percent nuclear boiling rate. This cycle yields a gross generator output of approximately 1700 MWe with a thermal reactor output of 4960 MW.

Turbine bypass system

The turbine bypass system (TBP) provides the capability to discharge main steam from the reactor directly to the condenser to minimize step load reduction transient effects on the reactor coolant system. The TBP is also used to discharge main steam during reactor hot standby and cooldown operations. The TBP consists of valve chest that is connected to the main steam lines upstream of the turbine stop valves, and dump lines that separately connect each bypass valve outlet to the condenser shell.

Main condenser

The main condenser, which does not serve or support any safety function and has no safety design basis, is a three-shell type deaerating condenser. During plant operation, steam expanding through the low-pressure turbines is directed downward into the main condenser and condensed. The main condenser also serves as a heat sink for the turbine bypass system, emergency and high level feedwater heater and drain tank dumps, and various other startup drains and relief valve discharges.

Each condenser shell has two tube bundles. Circulating water flows through the three shells. The main condenser is located in the turbine building in pits below the operating floor and is supported by the turbine building basemat.

6.2 Condensate and feedwater systems

The condensate and feedwater systems are designed to provide a dependable supply of high-quality feedwater to the reactor at the required flow, pressure, and temperature. The condensate pumps take the deaerated condensate from the main condenser hotwell and deliver it through the steam jet air ejector condenser, the gland steam condenser, the condensate demineralizer, and through three parallel strings of low pressure feedwater heaters to the reactor feed pump section. The reactor feed pumps discharge through two stages of high-pressure heater (two parallel strings) to the reactor.

6.3 Auxiliary systems

The turbine building cooling water system (TBCW), which is a non-safety related system, removes heat from the auxiliary equipment in the turbine building and rejects this heat to the turbine building service water system (TBSW) system.

Electrical and I&C systems

Basically the high-reliability design, which is comparable to that of the ABWR, is adopted for the auxiliary electrical power supply systems of the ABWR-II. However, there are three major differences between that of ABWR and that of ABWR-II:

1. The auxiliary normal electrical power supply systems of ABWR-II have partly applied the 11.5kV high voltage normal buses which feed power to the large capacity IMs such as RFPs of the 1700 MWe class plant. Those of ABWR have four 6.9kV high voltage buses.
2. The on-site emergency power supplies of ABWR-II are composed of two gas turbine driven generators and two diesel driven generators. Those of ABWR are composed of three diesel driven generators
3. The number of the safety-related buses of ABWR-II is four. That of ABWR is three.

7.1 Operational power supply systems

Description of major design features of the ABWR-II included in this category is outlined as a part of the paragraph 7. Electrical and I&C systems. The other design features except those mentioned in the paragraph 7 Electrical and I&C systems are basically the same as those of the ABWR.

7.2 Safety-related electrical systems

Description of major design features of the ABWR-II included in this category is outlined as a part of the paragraph 7. The DC power supplies of ABWR-II are also four divisions.

7.3 Instrumentation and control systems

Basically, a high-reliability design comparable to that of ABWRs is adopted for the ABWR-II. That is, integrated digitalized system is applied in the I&C of ABWR-II. It should be noted, however, that some of the instrumentation and control systems have been more sophisticated and highly reliable from the viewpoint of its practicality and effectiveness as described below:

(1) Advanced man-machine interface

On investigation of regular outage maintenance of the preceding ABWRs (K-6/7), some improvements to the ABWR plant man-machine interface have been suggested.

For example, the ABWR panels are good enough for start-up, shut-down and usual operation, however, as for regular outage some improvement needs have been mentioned, and shall be applied to the ABWR-II plant man-machine interface:

a. Efficiency of assistant operator and inspection team;
b. Visualization of inspection and maintenance status;
c. Judgment support of work conditions;
d. Efficiency of system inspection; and,
e. Improvement to I&C maintainability.

(2) Advanced control system

A symptom based transient mitigation method for feedwater system failure has been developed. This method will be able to monitor plant primary parameters and control other systems, if necessary, as the top plant control system in order to avoid unnecessary plant shut-down, controlling reactor water level.

This representative system is the automatic power output adjustable device. This is the system that outputs operating signals of the recirculation flow and control rods, etc. and controls the reactor power in case of feedwater system failure, based on information on the core status and the control rods.

(3) Transient mitigation system

This system has been developed to reduce DMCPR for the most severe transient as below:

a. Power load shut down bypass valve failure (DMinimum Critical Power Ratio (MCPR)=0.13)
b. Loss of feedwater heating (DMCPR=0.14)

To reduce these DMCPR, this system is equipped with the function to open the relief valve as soon as detecting the power load shut down bypass valve failure and also to activate the selected control rods insert or RIP run back signals as soon as detecting loss of feedwater heating from a change of feedwater temperature.

As a result of applying this system, DMCPR can be reduced to 0.08 for item (a) and 0.10 for item (b).

(4) Gamma thermometer

Gamma thermometer is activated by the principle that gamma thermometer rod shall be heated with nuclear fission and generated gamma rays. Generated heat is proportioned with surrounding fuel rods power density and flows to the coolant residual heat removal systems along the heat conducting pass. This heat along the conducting pass shall be measured by the thermo-couple and its signal is a result of the proportion to the core power output.

This system is applied for a correcting system of the core power.

(5) Advanced sensor technology

At first, the following investigations on the present process instrumentation methods of the nuclear plant and study on the application of ray technology based sensing or transmission method have been done:

a. Investigation on the present sensor
b. Investigation on the field bass method
c. Investigation on the ray fiber dyne method
d. Investigation on ray sensing instrumentation

As a result, ray fiber instrumentation system has been selected for ABWR-II design.

7.4 I&C design concepts, including control room

Description of major design features of the ABWR-II included in this category is outlined as a part of the paragraph 7.3 Instrumentation and control systems. The other design features except those mentioned in the paragraph 7.3 are basically the same as those of the ABWR.

7.5 Reactor protection and other I&C safety systems

Description of major design features of the ABWR-II included in this category is outlined as a part of paragraph 7.3. The other design features except those mentioned in the paragraph 7.3 are basically the same as those of the ABWR.

Spent fuel and waste management

No information provided.

Plant layout

9.1 Building and structures, including plot plan

No information provided.

9.2 Reactor building

Reactor building layout incorporates the following current reference design concepts:

- Rationalized 4 division RHR
- Passive heat removal systems
- RCIC with a generator
- Rationalized CR/CRD by function
- Gas turbine generators in addition to diesel generators

Since the reactor building layout for the current ABWR is highly optimized, the reference ABWR-II reactor building layout was modified in order to accommodate the above features and to reduce construction costs.

The ABWR-II reactor building is a steel plate reinforced concrete structure. The integrated reactor building and cylindrical containment structure is adopted to improve constructability, to reduce construction costs and the construction schedule. A secondary containment surrounds the primary containment and provides a second containment function including a standby gas treatment system.

Key features of the ABWR-II reactor building layout include:

a. Suppress the increment of building volume due to power up-grade from current ABWR,
b. Keep the same operability and maintainability as current ABWR,
c. Keep simple structural shapes to improve constructability (reduce costs and schedule).

Two types of reactor building layout complying with slightly different Primary Containment Vessel (PCV) configurations were studied. The volumes of these ABWR-II reactor buildings were approximately 102%-104% of the current ABWR (18%-20% less than the current ABWR at per power ratio) while keeping the advantages of the current ABWR such as operability and maintainability, etc.

9.3 Containment

The primary containment vessel configuration incorporates the following current reference design concepts:

- Large capacity SRV
- Low pressure drop MSIV

The Modified ABWR containment is based on the reinforced concrete containment vessel of the proven ABWR. This containment has the following features:

a. Reinforced concrete will provide the strength necessary to withstand the pressure, and an interior steel liner will ensure the required air tightness;
b. Cylindrical structure is integrated with reactor building;
c. The top slab serves as a portion of the spent fuel pool and dryer-separator pool;
d. Vessel features a horizontal vent, access tunnel in the suppression chamber and rigid diaphragm floor.

Reflecting the consideration of the thermal output level, the diameter and height of the containment vessel were carefully reviewed from a safety-design point of view, so the height of the containment has been increased. (See Figure 9.1)

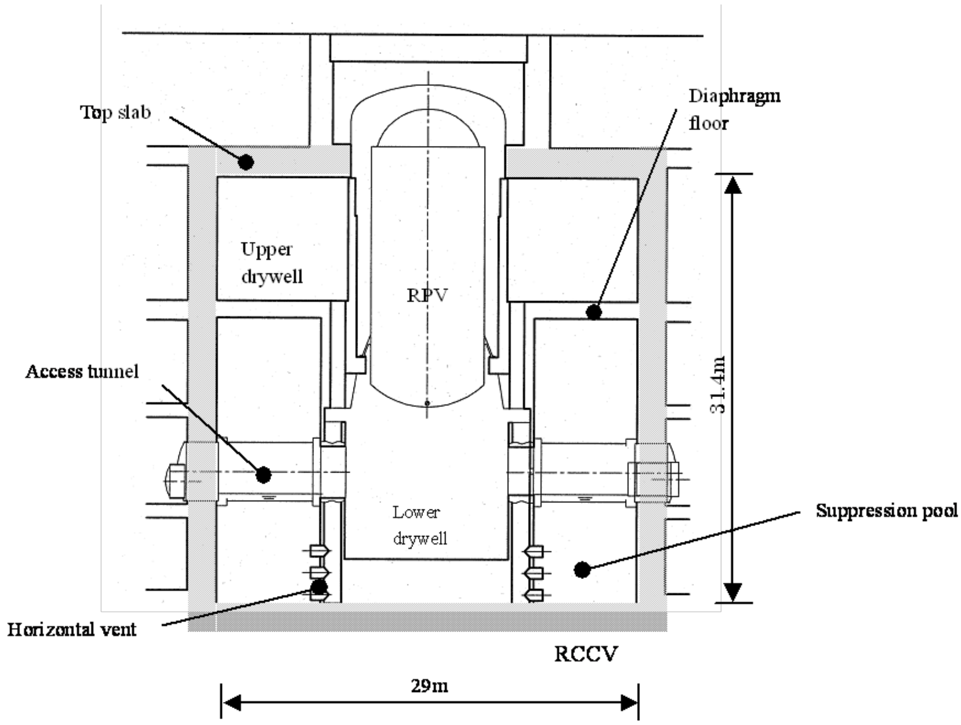


FIG. 9.1. Section View of Modified ABWR PCV

Plant performance

In planning for a future reactor, it is indispensable to set a cost target for power generation. For ABWR-II, the challenging target of 30% reduction in power generation cost from that of a standardized ABWR was set. Nuclear power plants have relatively high construction cost and low running cost compared to fossil power plants. Therefore, capital cost reduction by design has been carefully looked into in addition to operation and maintenance (O&M) cost reduction.

The following are design considerations to improve ABWR-II economics.

10.1 Design simplification

When the Phase I program started, the ABWR-II plant power output was set at 1350 MWe, the same as ABWR.

During Phase II, when the need for cost reduction increased, the reference output was increased to 1500 MWe to obtain larger merit by economies of scale. In Phase III, it became apparent the target of 20% power generation cost reduction was so challenging that further output increase should be required. The output was again increased to 1700 MWe as a reference.

This 1700 MWe output was decided considering compatibility with Japanese grid capacity and manufacturability for components such as reactor pressure vessels and generators. The larger output would be suitable also for future replacement of old plants because of better efficiency in using limited site area and common facilities.

The new type MSIV for ABWR-II is now under development. The bore diameter of the MSIV is increased and the center of gravity of its driving mechanism is lowered. This diameter increase is not simply an enlargement from ABWR but is optimized in such a way that the pressure loss will be decreased from ABWR to increase efficiency. The lowered center of gravity of the driving mechanism by relocating springs and oil damper will contribute to improvement of seismic capability. This design makes it possible to simplify supporting rigs against seismic effects.

For the ABWR-II, an increase of discharge capacity and simplification of valve structure were considered. In order to decrease the number of SRVs, the discharge capacity per SRV is increased by 70% to 680 t/h (nominal) from ABWR's 395 t/h with an increased throat diameter and increased coil spring diameter. At the same time, the structure of SRV is simplified by integrating an air cylinder into the SRV main body.

In order to have enough flexibility for future higher burnup and longer cycle operation and to reduce components and shorten the refueling outage time, a 1.5 times larger fuel bundle and K-lattice were selected for the reference design. Though power output of ABWR-II is increased to 1700 MWe from 1356 MWe, the number of fuel bundles is decreased to 424 from 872, and the number of control rods is decreased to 197 from 205.

In the CRD design of ABWR-II, an improvement in the connection between motor and CRD shaft was made. The improved mechanism is called a magnetic coupling. The magnetic coupling can transmit torque between the CRD motor and the CRD shaft through the pressure boundary without a penetration instead of having a seal around a penetrating shaft. This improvement eliminates the sealing parts where inspection and maintenance are most necessary. In addition, the improved HCU for CRD, which is under development, will be adopted for ABWR-II. In case of conventional HCUs, 99 HCUs are required for 197 CRs. By applying the improved ones, 65 improved HCUs and one conventional HCU are sufficient.

Optimization was made in the RHR system together with the RCW and RSW systems. Taking into consideration that the passive heat removal systems of ABWR-II can be considered as backup, the basic configuration of RCW is two divisions instead of the three in ABWR. This two-division configuration is expected to reduce equipment cost for RCW that has relatively large amount of materials especially for piping.

Flammable gas control in the containment is performed by the combination of inerting and passive autocatalytic recombiner (PAR), which has advantages both of safety (automatic startup and passive operation) and economy (low cost, flexible layout and easy maintenance).

The ABWR-II reactor building is a steel plate reinforced concrete structure. The integrated reactor building and cylindrical containment structure is adopted to improve constructability, to reduce construction costs and the construction period. Two types of reactor building layout complying with slightly different Primary Containment Vessel (PCV) configurations were studied. The volume of these ABWR-II reactor building were approximately 102%-104% relative to the current ABWR (meaning 18%-20% less than the current ABWR at per power ratio) while keeping the advantages of the current ABWR such as operability and maintainability, etc.

ABWR-II plant system features are summarized in Figure 10.1.

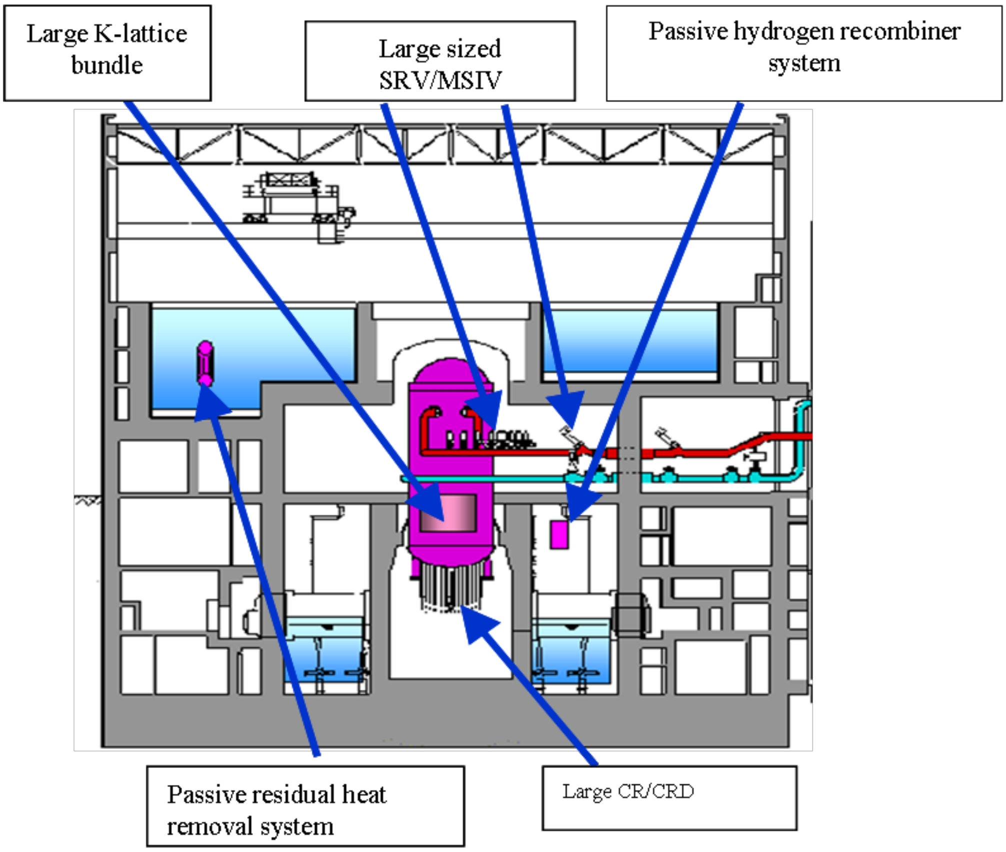


FIG. 10.1. Features of ABWR-II Plant System

10.2 Operational flexibility improvement

Targeted operating cycle length and average discharge burnup of the ABWR-II core are 18 effective full power months (EFPM) and 60 GWd/t, respectively. The recirculation flow control range of 20% is achieved. In the BWR, excess reactivity is controlled by burnable absorbers, CRs and recirculation flow. Recirculation flow can control the reactivity through the void fraction, which changes the neutron spectrum in the core. In order to reduce use of CRs under ordinary operation and improve the neutron economy, it is desirable to enlarge the controllable reactivity using recirculation flow. The reference core design has a more negative void coefficient due to larger fuel inventory, and reactivity control capability with core flow becomes larger than current core design. In order to enlarge the reactivity change due to unit recirculation flow change further, the SSR has been developed as an option to be used instead of the water rod, in which the water level develops naturally during operation and changes according to the recirculation flow rate through the channel. In this case, it is expected that fuel cycle cost will be reduced and that CRs need not be replaced periodically, contributing to additional refueling time reduction.

For RHR, RSW and active components in RCW, in total, make up a four-division configuration that facilitates on-line maintenance and increases reliability and safety. As to emergency power sources for active components in RHR/RCW/RSW systems, a four-division configuration consisting of two diesel generators and two gas turbine generators is applied to increase diversity and to facilitate maintenance. On-line maintenance will be applied to the diesel generators. The gas turbine generators are expected to be maintenance free. Therefore, increased reliability and a reduced maintenance outage period will be achieved by this optimized division combination of two and four.

A new feature of ABWR-II design is adoption of passive safety systems. These systems not only cover beyond DBA condition, but also provide in-depth heat removal backup for RHR, and practically eliminate necessity of containment venting before and after core damage as a means of overpressure protection. Flammable gas control in the containment is performed by the combination of inerting and passive autocatalytic recombiner (PAR), which has advantages both of safety (automatic startup and passive operation) and economy (low cost, flexible layout and easy maintenance).

10.3 Cost reduction of equipment and structures

Improving equipment itself and/or decreasing number of components can mean cost reduction. In addition, improvement of plant efficiency contributes to cost reduction. Major cost reductions are represented by the following:

- Low pressure drop type separator (Refer to Ref. [1])
- Large capacity and simplified mechanism SRV
- Low pressure drop MSIV improving plant efficiency
- Simplified mechanism CRD by magnet coupling
- Horizontal type heat exchanger for PCCS

The ABWR-II reactor building is a steel plate reinforced concrete structure. The integrated reactor building and cylindrical containment structure is adopted to improve constructability, to reduce construction costs and the construction schedule. A secondary containment surrounds the primary containment and provides a second containment function including a standby gas treatment system.

10.4 Reduction of construction period

ABWR achieved construction schedule of 37 months (from first concrete work to fuel loading) at the Kashiwazaki-Kariwa first-of-a-kind plant. The construction schedule target of ABWR-II is 29.5 months. This remarkable reduction of construction schedule is expected by the design organization to be achieved by the following procedures including modified ABWR construction technologies:

- Large scale modularization
- Open-top installation method
- Rationalization of testing
- Integrated module of both mechanical and civil
- SC structure building
- Extra large scale crane
- Expansion of the application of automatic welding machinery

10.5 Scope reduction of the maintenance during operation and outages

The large 1.5 K-lattice core arrangement enables a decrease in the number of fuel bundles and CRDs. These contribute to additional refueling time reduction and CRDs maintenance time reduction.

The magnetic coupling can transmit torque between the CRD motor and the CRD shaft through the pressure boundary without a penetration instead of having a seal around a penetrating shaft. This improvement eliminates the sealing parts where inspection and maintenance are most necessary, and also the seal detection system that requires monitoring during operation, thus making CRDs maintenance free, contributing to maintenance cost reduction.

For RHR, RSW and active components in RCW, in total, make up a four-division configuration that facilitates on-line maintenance and increases reliability and safety. As to emergency power sources for active components in RHR/RCW/RSW systems, a four-division configuration consisting of two diesel generators and two gas turbine generators is applied to increase diversity and to facilitate maintenance. On-line maintenance will be applied to the diesel generators. The gas turbine generators are expected to be maintenance free. Therefore, increased reliability and a reduced maintenance outage period will be achieved by this optimized division combination of two and four. This on-line maintenance changes the scope of the maintenance during operation and outages.

Passive systems such as PCCS and PAR are applied for ABWR-II, reducing maintenance work typically associated with active systems.

10.6 Making the maintenance easier and with lower radiation exposure

The large 1.5 K-lattice core arrangement enables a decrease in the number of fuel bundles and CRDs. These contribute to additional refueling time reduction and CRDs maintenance time reduction. Therefore the large 1.5 K-lattice core arrangement make the maintenance easier and with lower radiation exposure.

Simplified mechanism and large capacity type SRV is applied for ABWR-II. This SRV contribute to make the maintenance easier and with lower radiation exposure.

Magnet coupling CRD eliminates the sealing parts where inspection and maintenance are most necessary, and also the seal detection system that requires monitoring during operation, thus making CRDs maintenance free, contributing to easier maintenance and with lower radiation exposure.

Taking into consideration that the passive heat removal systems of ABWR-II can be counted as backup, the basic configuration of RCW is two divisions instead of the three in ABWR. This two-division configuration is expected to make maintenance easier and lower radiation exposure.

10.7 Increasing the capacity factor

Refueling outage period reduction is a major factor in power generation cost reduction because it improves the capacity factor. In the ABWR-II program, outage period reduction has been approached from two aspects:

- Design for maintenance reduction or on-line maintenance
- Expected future deregulation

Examples of design for maintenance reduction or on-line maintenance are:

- Reduced number of fuel bundles
- CRD boundary penetration shafts elimination
- Reduced number and simplification of SRVs
- Four-division RSW system configuration
- Four-division emergency power sources

As to deregulation, maintenance interval extension and rationalization of regulatory audit schedule and test items was taken into consideration.

After having checked feasibility of a 30-day refueling outage period, the current target is a further reduction to 20 days. Considering an operation cycle of 18 months, the capacity factor with a 20-day refueling outage will be 96%. Since this 20-day refueling period considers minimum maintenance work, there will be some longer outages once in a while through the plant lifetime. The average capacity factor through the plant lifetime would be expected to be more than 90%.

10.8 Reduction of the power generating cost

There are several ways to reduce power generation cost for new nuclear power plants. Figure 10.2 shows the breakdown of the nuclear power generation costs as reported by the Agency of Natural Resources and Energy in Japan. The power generation cost is largely divided into three categories: capital cost, operation and maintenance (O & M) costs and fuel cost. Since the capital cost accounts for 39% of the total, its reduction would seem to be the most effective way. However, simple elimination of components is not a good choice. For example, engineering safety features are around 6% of the total cost, thus the impact on total cost reduction is rather small even if they could be eliminated completely. Therefore, increasing plant output while minimizing impact on the plant systems is the best way to reduce the capital cost. Moreover, O & M costs are almost constant regardless of the plant output. In short, two dominant factors, which occupy 71% of the power generation costs, can be directly reduced by the plant output increase.

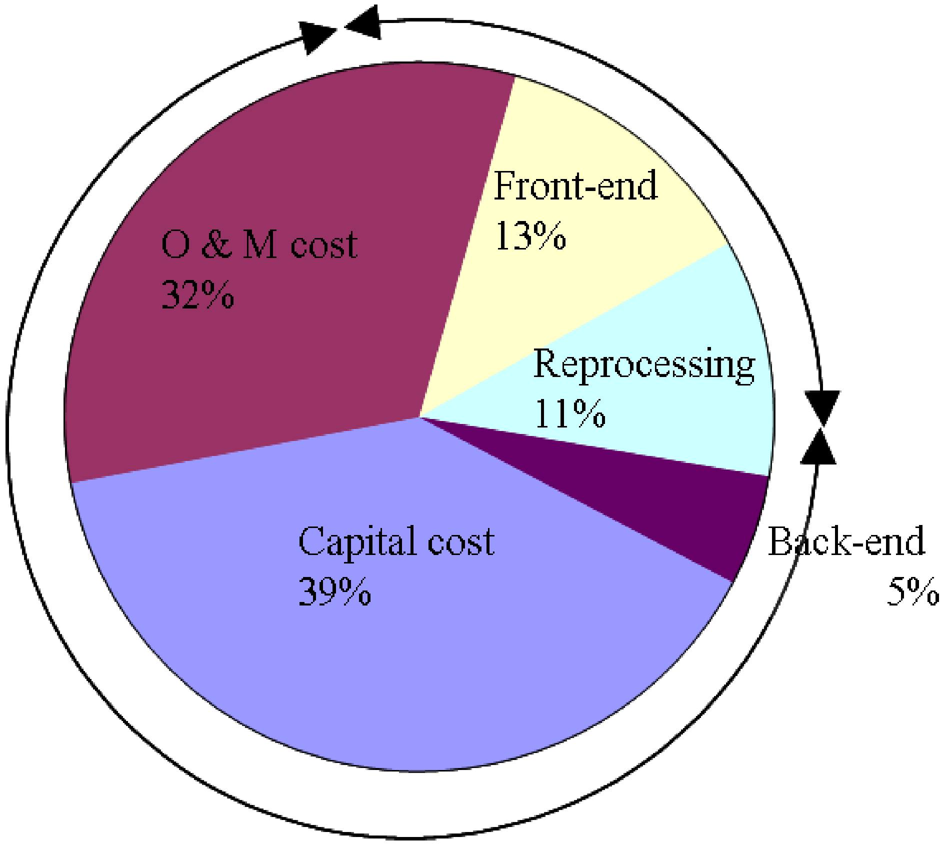


FIG. 10.2. Power Generation Cost Breakdown

Figure 10.3 summarizes cost reduction effects on the power generation costs by introducing the ABWR-II. Based on the current reference design concept as shown above, the plant construction cost for a 1700 MWe ABWR-II is estimated to be almost the same as that of a 1350 MWe ABWR. Therefore, the 26% power increase from 1350MWe to 1700MWe directly affects the capital cost and is expected to reduce it by approximately 20%. Design simplification, operation flexibility improvement and increasing the capacity factor make it possible to reduce the O&M costs remarkably. About 20% reduction is also expected for O&M costs. Burn-up extension up to 60 GWd/t by incorporating the SSR allows fuel cycle cost to be reduced by 7%.

Overall cost reduction for power generation is estimated to be about 15-20% against the ABWR.

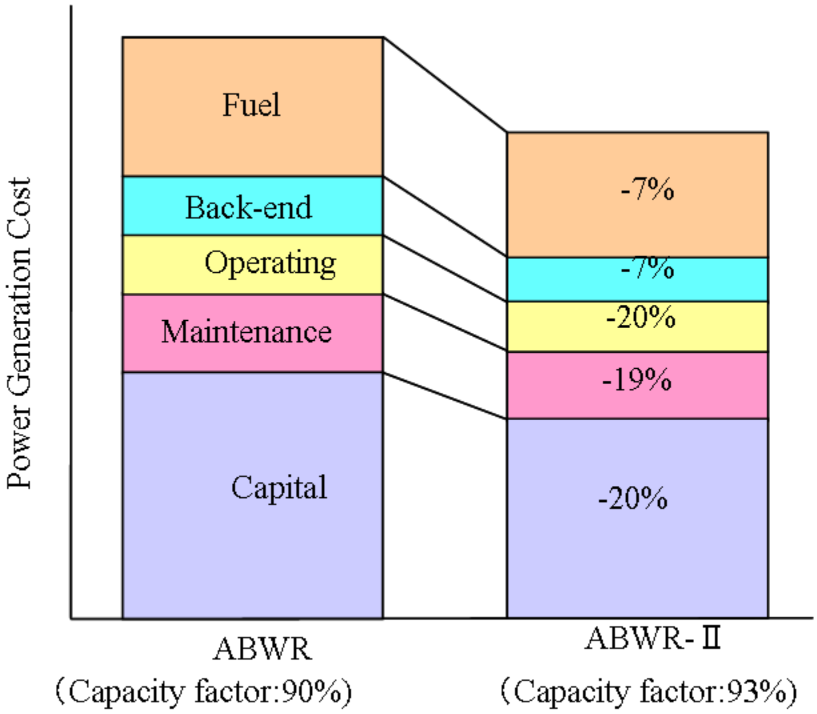


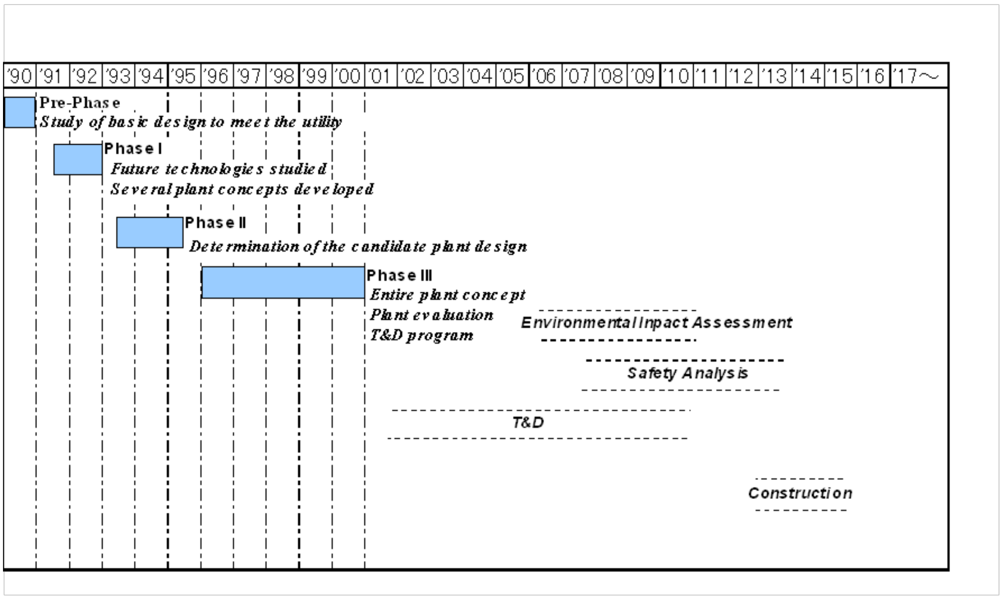
FIG. 10.3. Cost Reduction Effect of ABWR-II

Development status of technologies relevant to the NPP

No information provided.

Deployment status and planned schedule

The ABWR-II development project was initiated over a decade ago and has completed three phases to date. In Phase I (1991-92), basic design requirements were discussed and several plant concepts were studied. In Phase II (1993-95), key design features were selected in order to establish a reference reactor concept. In Phase III (1996-2000), based on the reference reactor concept, modifications and improvements were made to fulfil the design requirements. And in the present, various testing programmes are performed or planned to consolidate their feasibilities and to find further room for improvements. The commercial introduction of ABWR-II is now set by the latter half of the 2010s when replacements of older nuclear power plants are expected to start.


'90 '91 '92 '93 '94 '95 '96 '97 '98 '99 '00 '01 '02 '03 '04 '05 '06 '07 '08 '09 '10 '11 '12 '13 '14 '15 '16 '17～
Pre-Phase
Study of basic design to meet the utility
Phase I
Future technologies studied
Several plant concepts developed
Phase II
Determination of the candidate plant design
Phase III
Entire plant concept
Plant evaluation
T&D program
Environmental Inpact Assessment
Safety Analysis
T&D
Construction

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Technical data

General plant data

Reactor thermal output	4960 MWth
Power plant output, gross	1717 MWe
Power plant output, net	1638 MWe
Power plant efficiency, net	33.0 %

Primary coolant material	Light Water
Moderator material	Light water
Thermodynamic cycle	Rankine
Type of cycle	Direct

Nuclear steam supply system

Steam flow rate at nominal conditions	2681 Kg/s
Steam pressure	7.17 MPa(a)
Steam temperature	287.8 °C
Feedwater flow rate at nominal conditions	2677 Kg/s
Feedwater temperature	215.5 °C

Reactor coolant system

Primary coolant flow rate	15667 Kg/s
Reactor operating pressure	7.17 MPa(a)
Core coolant inlet temperature	277 °C
Core coolant outlet temperature	288 °C
Mean temperature rise across core	11 °C

Reactor core

Active core height	3.71 m
Equivalent core diameter	5.41 m
Average fuel power density	26.1 KW/KgU
Average core power density	58.1 MW/m^3
Fuel material	Sintered UO2
Outer diameter of fuel rods	10.3 mm
Rod array of a fuel assembly	(8x8)x4
Lattice geometry	Square
Number of fuel assemblies	424
Enrichment of reload fuel at equilibrium core	5.2 Weight %
Fuel cycle length	18 Months
Average discharge burnup of fuel	60 MWd/Kg
Burnable absorber (strategy/material)	Gd2O3-UO2
Control rod absorber material	B4C and Hf

Soluble neutron absorber	Boron

Reactor pressure vessel

Inner diameter of cylindrical shell	7450 mm
Wall thickness of cylindrical shell	190 mm
Design pressure	5.62 MPa(a)
Design temperature	302 °C
Base material	Carbon Steel
Total height, inside	21300 mm

Reactor coolant pump (Primary circulation System)

Pump Type	Vertical internal, Variable Speed, Single Stage, Wet Motor
Number of pumps	10
Pump speed	1500 rpm
Head at rated conditions	34.7 m
Flow at rated conditions	1.725 m^3/s

Primary containment

Type	Pressure-suppresion reinforced concrete
Overall form (spherical/cylindrical)	Cylindrical
Dimensions - diameter	29 m
Dimensions - height	31.4 m
Design pressure	0.310 MPa
Design temperature	171 °C

Residual heat removal systems

Active/passive systems	Passive

Turbine

Type of turbines	Six flow, Tandem compound, Single reheat
Number of turbine sections per unit (e.g. HP/MP/LP)	1/0/3

Generator

Type	Three-phase, turbo generator
Active power	1700 MW

Condenser

Type	Shell type