Status report 97 - Advanced Boiling Water Reactor (ABWR)

Overview

Full name	Advanced Boiling Water Reactor
Acronym	ABWR
Reactor type	Boiling Water reactor (BWR)
Coolant	Light Water
Moderator	Light water
Neutron spectrum	Thermal Neutrons
Thermal capacity	3926.00 MWth
Gross Electrical capacity	1420.00 MWe
Design status	In Operation
Designers	GE-Hitachi
Last update	21-07-2011

Description

Introduction

1. INTRODUCTION

The design of the advanced boiling water reactor (ABWR) represents a complete design for a nominal 1350 MWe power plant. The inclusion of such features as reactor internal pumps, fine motion control rod drives, multiplexed digital fiber-optic control systems, and an advanced control room are examples of the type of advancements over previous designs that have been incorporated to meet the ABWR objectives.

The ABWR design objectives include: 60 year plant life from full power operating license date, 87% or greater plant availability, less than one unplanned scram per year, 24 month refuelling interval, personnel radiation exposure limit of 100 man-rem/year, core damage frequency of less than 10^{-5}/ reactor year, limiting significant release frequency to 10^{-6}/reactor year, and reduced radwaste generation.

The principal design criteria governing the ABWR standard plant encompass two basic categories of requirements: those related to either a power generation function or a safety related function.

General power generation design criteria

The plant is designed to produce electricity from a turbine generator unit using steam generated in the reactor.

Heat removal systems are designed with sufficient capacity and operational adequacy to remove heat generated in the reactor core for the full range of normal operational conditions and abnormal operational transients. Backup heat removal systems are designed to remove decay heat generated in the core under circumstances where the normal heat removal systems become inoperative. The capacity of such systems is adequate to prevent fuel cladding damage.

The fuel cladding, in conjunction with other plant systems, is designed to retain its integrity so that the consequences of any equipment failures are within acceptable limits throughout the range of normal operational conditions and abnormal operational transients for the design life of the fuel.

Control equipment is designed to allow the reactor to respond automatically to load changes and abnormal operational transients. Reactor power level is manually controllable.

Interlocks or other automatic equipment are designed as backup to procedural control to avoid conditions requiring the functioning of safety related systems or engineered safety features.

General safety design criteria

The plant is designed, fabricated, erected and operated in such a way that the release of radioactive material to the environment does not exceed the limits and guideline values of applicable government regulations pertaining to the release of radioactive materials for normal operations, for abnormal transients and for accidents.

The reactor core is designed so that its nuclear characteristics counteract a power transient. The reactor is designed so that there is no tendency for divergent oscillation of any operating characteristics considering the interaction of the reactor with other appropriate plant systems.

Safety related systems and engineered safety features function to ensure that no damage to the reactor coolant pressure boundary results from internal pressures caused by abnormal operational transients and accidents. Where positive, precise action is immediately required in response to abnormal operational transients and accidents, such action is automatic and requires no decision or manipulation of controls by plant operations personnel.

The design of safety related systems, components and structures includes allowances for natural environmental disturbances such as earthquakes, floods, and storms at the plant site.

Standby electrical power sources have sufficient capacity to power all safety-related systems requiring electrical power concurrently. Standby electrical power sources are designed to allow prompt reactor shutdown and removal of decay heat under circumstances where normal auxiliary power is not available.

A containment is provided that completely encloses the reactor systems, drywell, and pressure suppression "wetwell" chambers. The containment employs the pressure suppression concept.

A safety envelope is provided that basically encloses the containment, with the exception of the areas above the containment top slab and drywell head. The containment and safety envelope, in conjunction with other safety related features, limit radiological effects of design basis accidents to less than the prescribed acceptable limits. The reactor building surrounds the containment/safety envelope and serves as a secondary containment.

Provisions are made for removing energy from the containment as necessary to maintain the integrity of the containment system following accidents that release energy to the containment.

Emergency core cooling is designed to limit fuel cladding temperature to less than the limits of the U.S. NRC Regulation 10CFR50.46 (2200°F or 1204°C) in the event of a design basis loss of coolant accident (LOCA). The emergency core cooling is designed for continuity of core cooling over the complete range of postulated break sizes in the reactor coolant pressure boundary piping. Emergency core cooling is initiated automatically when required regardless of the availability of off site power supplies and the normal generating system of the plant.

The control room is shielded against radiation so that continued occupancy under design basis accident conditions is possible. In the event that the control room becomes uninhabitable, it is possible to bring the reactor from power range operation to cold shutdown conditions by utilizing alternative controls and equipment that are available outside the control room.

Fuel handling and storage facilities are designed to prevent inadvertent criticality and to maintain shielding and cooling of spent fuel as necessary to meet operating and off-site dose constraints.

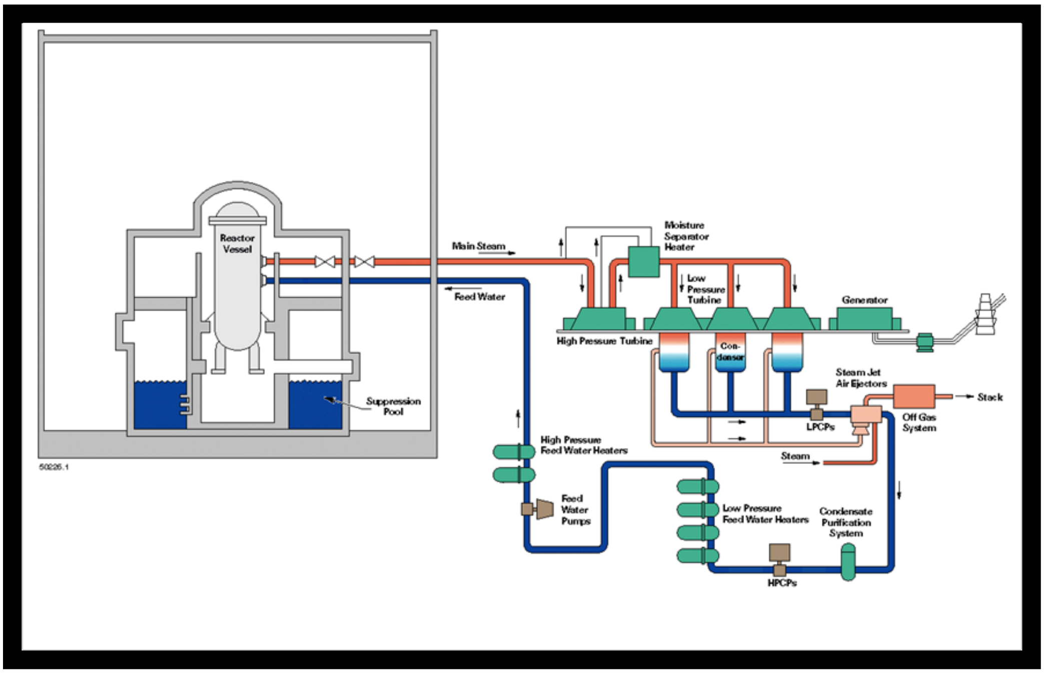

FIG. 1.1. ABWR - steam cycle

Description of the nuclear systems

2.1 Primary circuit and its main characteristics

Figure 1.1 illustrates the ABWR steam cycle. The primary functions of the 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,
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, and
5. with the exception of monitoring the neutron flux, to provide the instrumentation necessary for monitoring conditions in the RPV such as RPV pressure, metal temperature, and water level instrumentation.

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

The main steam line flow limiter, a flow restricting venturi built into the RPV MSL nozzle of each of the four main steam lines, limits the coolant blowdown rate from the reactor vessel to a (choke) flow rate equal to or less than 200% of rated steam flow at 7.07 MPa (1025 psig) upstream gauge pressure in the event 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 inner MSIV in the containment and one outer 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 nuclear pressure relief system consists of safety/relief valves (SRVs) located on the main steam lines (MSLs) between the RPV and the inboard main steam line isolation valve. There are 18 SRVs distributed on the four MSLs. The SRVs are designed to provide three main protection functions: overpressure safety, overpressure relief, and depressurization operation.

The automatic depressurization subsystem (ADS) consists of the eight 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.

2.2 Reactor core and fuel design

The ABWR core configuration consists of 872 bundles. The rated core power is 3926 MWt, which corresponds to a 50.6 kW/l power density. The lower power density results in improved fuel cycle costs and greater maneuverability. Since the ABWR utilizes reactor internal pumps (RIPs) to control the recirculation flow through the core, the reactivity control is maintained by a combination of changes in core flow, control rod position and by the inclusion of burnable absorbers in the fuel.

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) subsystem.

The FMCRDs (Figure 2.1 shows a cross-section) are designed to provide electric-motor-driven positioning for normal insertion and withdrawal of the control rods and hydraulic-powered rapid control rod insertion (scram) in response to manual or automatic signals from the reactor protection system (RPS). In addition to hydraulic-powered scram, the FMCRDs also provide electric-motor-driven run-in of all control rods as a path to rod insertion that is diverse from the hydraulic powered scram. The hydraulic power required for scram is provided by high-pressure water stored in the individual HCUs. The HCUs also provide the flow path for purge water to the associated drives during normal operation. The CRDH subsystem supplies high-pressure demineralized water, which is regulated and distributed to provide charging of the HCU scram accumulators, purge water flow to the FMCRDs, and backup makeup water to the RPV when the feedwater flow is not available.

There are 205 FMCRDs mounted in housings welded into the RPV bottom head. Each FMCRD has a movable hollow piston tube that is coupled at its upper end, inside the reactor vessel, to the bottom of a control rod. The piston is designed such that it can be moved up or down, both in fine increments and continuously over its entire range, by a ball nut and ball screw driven at a nominal speed of 30 mm/s by the electric stepper motor.

In response to a scram signal, the piston rapidly inserts the control rod into the core hydraulically using stored energy in the HCU scram accumulator. The FMCRD design includes an electro-mechanical brake on the motor drive shaft and a ball check valve at the point of connection with the scram inlet line. These features prevent control rod ejection in the event of a failure of the scram insert line. There are 103 HCUs, each of which provides sufficient water stored at high pressure in a pre-charged accumulator to scram two FMCRDs at any reactor pressure.

In Japan, a sealless FMCRD, a new type of FMCRD, has been developed and deployed in Hamaoka Unit-5, which commenced commercial operation in January 2005. Its basic design is shown in Figure 2.1, on the right. The purpose of the sealless FMCRD is to eliminate FMCRD shaft penetration through the pressure boundary, utilizing a magnet coupling to transmit drive force from the electric motor. This new system eliminates the ground packing and leak detection system of the conventional FMCRD, and enhances reliability and maintainability. In addition, the FMCRD's step width requirement has been mitigated by improvements in fuel design, and a change in the motor-drive system from a stepper motor and inverter power source to an induction motor with AC power source and contactor.

2.3 Fuel handling and transfer systems

The reactor building is supplied with a refuelling machine for fuel movement and servicing plus an auxiliary platform for servicing operations from the vessel flange level.

The refuelling machine is a gantry crane, which spans the reactor vessel and the storage pools on bedded tracks in the refuelling floor. A telescoping mast and grapple suspended from a trolley system is used to lift and orient fuel bundles for placement in the core and/or storage racks. Two auxiliary hoists, one main and one auxiliary monorail trolley-mounted, are provided for in-core servicing. Control of the machine is from an operator station on the refuelling floor.

A position indicating system and travel limit computer are provided to locate the grapple over the vessel core and prevent collision with pool obstacles. The mast grapple has a redundant load path so that no single component failure results in a fuel bundle drop. Interlocks on the machine: (1) prevent hoisting a fuel bundle over the vessel unless an all-control-rod-in permissive is present; (2) limit vertical travel of the fuel grapple to provide shielding over the grappled fuel during transit; and, (3) prevent lifting of fuel without grapple hook engagement and load engagement.

Storage racks are provided for the temporary and long-term storage of new and spent fuel and associated equipment. The new and spent fuel storage racks use the same configuration and prevent inadvertent criticality.

Racks provide storage for spent fuel in the spent fuel storage pool in the reactor building. New fuel, 40% of the reactor core, is stored in the new fuel storage vault in the reactor building. The racks are top loading, with fuel bail extended above the rack. The spent fuel racks have a minimum storage capacity of 270% of the reactor core, which is equivalent to a minimum of 2354 fuel storage positions. The new and spent fuel racks maintain a subcriticality of at least 5% Dk under dry or flooded conditions. The rack arrangement prevents accidental insertion of fuel assemblies between adjacent racks and allows flow to prevent the water from exceeding 100°C.

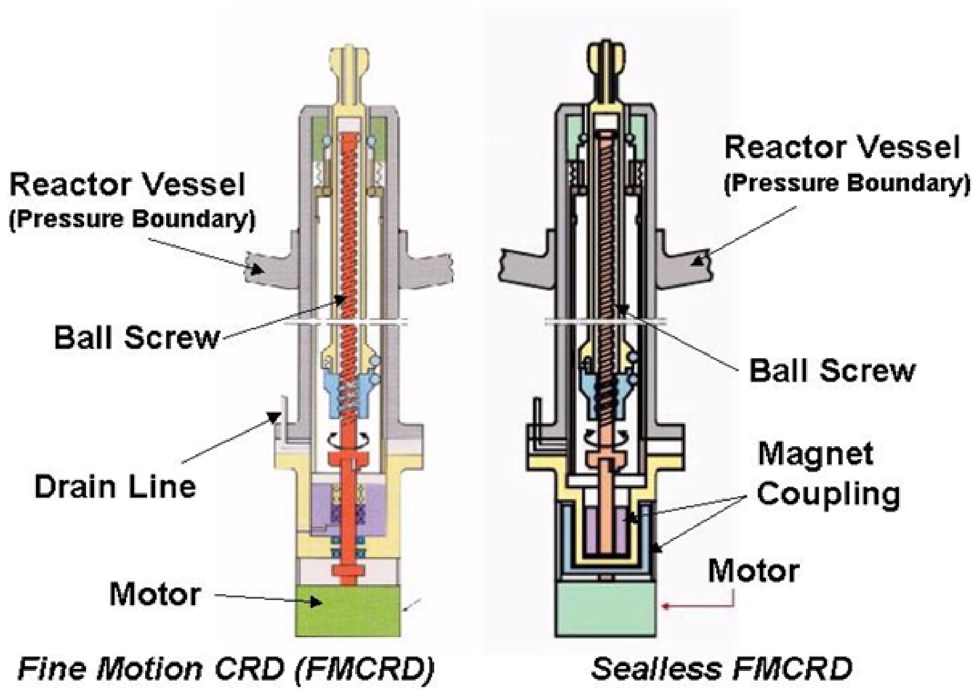

FIG. 2.1. Cross-section of fine motion control rod drive (FMCRD) and sealless FMCRD

2.4 Primary components

Reactor pressure vessel

The reactor pressure vessel (RPV) system consists of (1) the RPV and its appurtenances, supports and insulation, excluding the loose parts monitoring system, 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 (RCPB) 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 RCPB 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 RCPB 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.

The large RPV volume provides a large reserve of water above the core, which translates directly into a much longer period of time (compared to prior BWRs) before core uncovery is likely to occur as a result of feedwater flow interruption or a LOCA. This gives an extended period of time during which automatic systems or plant operators can re-establish reactor inventory control using any of several normal, non-safety-related systems capable of injecting water into the reactor. Timely initiation of these systems precludes the need for activation of emergency safety equipment. The large RPV volume also reduces the reactor pressurization rates that develop when the reactor is suddenly isolated from the normal heat sink, which eventually leads to actuation of the safety-relief valves.

Reactor internals

The ABWR RPV and internals are illustrated in Figure 2.2. The major reactor internal components in the RPV System are: (1) core support structures; and, (2) other reactor internals.

The Core support structures encompass: the shroud, shroud support and a portion of CRD housings inside the reactor internals RPV; core plate; top guide; fuel supports; 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 separators assembly and the steam dryer assembly.

Reactor recirculation pumps

The reactor recirculation system (RRS) 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.

By regulating the flow rate, the reactor power output can be regulated over an approximate range from 70 to 100% of rated output, without moving control rods. RIP performance is adequate to allow plant operation at 100% power with only 9 of the 10 pumps in operation.

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 cooling water system. Figure 2.3 illustrates a cross-section of a RIP.

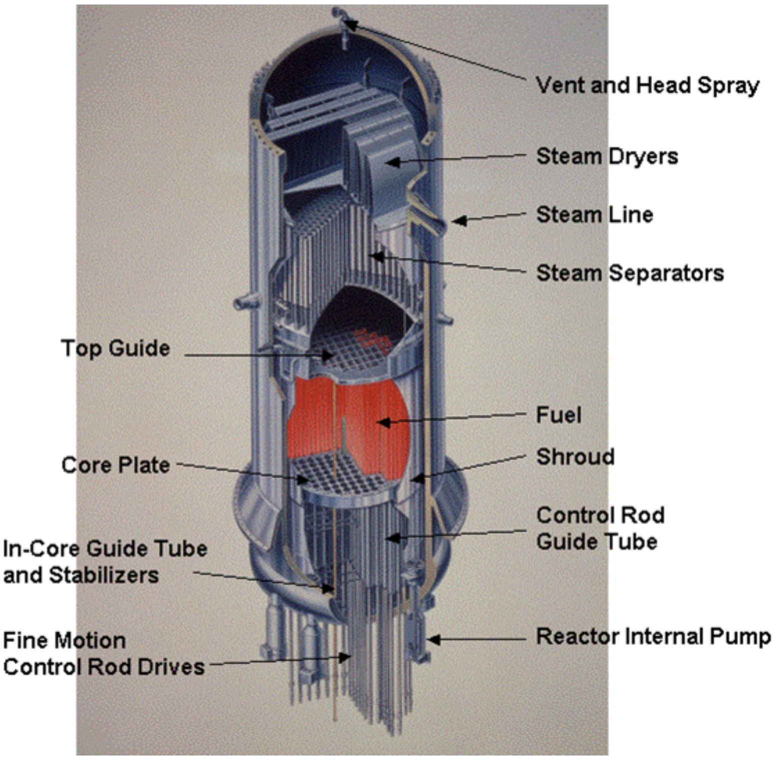

FIG. 2.2. ABWR – reactor pressure vessel and internals

2.5 Reactor auxiliary systems

The main auxiliary systems in the ABWR nuclear island consist of the reactor building cooling water (RBCW) system, the reactor water cleanup (RWCU) system, the fuel pool cooling and cleanup (FPCU) 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, chilled water, HVAC, equipment drain, floor drain and other systems that are basically the same as on past BWR plants and are not covered here since the designs are all well proven.

The RBCW system consists of piping, valves, pumps and heat exchangers that are used to provide cooling water to the various consumers in the nuclear island. The system is divided into three separate safety divisions, each with its own pump and heat exchanger, to provide cooling water to equipment in the three ECCS and RHR safety divisions. The RBCW system also provides cooling water to equipment in non-safety systems such as the RWCU, FPCU and other systems and equipment that require cooling water.

The RWCU heat exchangers are cooled by water from the plant service water or ultimate heat sink depending on unique site conditions.

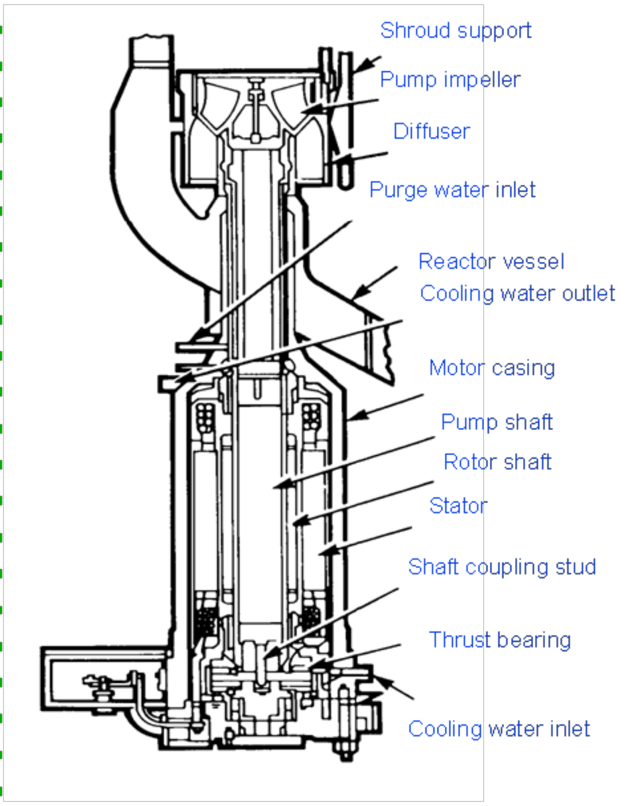

FIG. 2.3. Cross-section of reactor internal pump

The RWCU system consists of piping, valves, pumps, heat exchangers and filter demineralizers that are used to remove impurities from the reactor primary coolant water to maintain water quality within acceptable limits during the various plant operating modes. The RWCU design for ABWR is basically the same as on previous BWRs with the following exceptions: 1) the RWCU pumps are located downstream of the regenerative and non-regenerative heat exchangers to reduce the pump operating temperature and improve pump seal and bearing performance, and 2) two 1% capacity systems are used instead of only one 1% system, as found in previous BWRs.

The FPCU 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 various plant operating modes. The filter-demineralizer in the FPCU system is shared by the SPCU system for cleaning the suppression pool water. The FPCU and SPCU systems are basically the same as on previous BWRs.

2.6 Operating characteristics

The ABWR design incorporates extensive automation of the operator actions that are required during a normal plant startup, shutdown and power range maneuvers. The automation features adopted for the ABWR are designed for enhanced operability and improved capacity factor, relative to conventional BWR designs. However, the extent of automation implemented in the ABWR has been carefully selected to ensure that the primary control of plant operations remains with the operators. The operators remain fully cognizant of the plant status and can intervene in the operation at any time, if necessary.

The ABWR control room design provides the capability for a single operator to perform all required control and monitoring functions during normal plant operations as well as under emergency plant conditions. One man operation is possible due to implementation of several key design features: the wide display panel for overall plant monitoring; plant-level automation; system-level automation via sequence master control switches; the compact main control console design; and, implementation of operator guidance functions that display appropriate operating sequences on the main control panel CRTs. The role of the operator will primarily be one of monitoring the status of individual systems and the overall plant and the progress of automation sequences, rather than the traditional role of monitoring and controlling individual system equipment. However, to foster a team approach in plant operation and to maintain operator vigilance, the operating staff organization for the reference ABWR control room design is based upon having two operators normally stationed at the control console.

The incorporation of reactor internal pumps (RIPs) allows power changes of up to 30% of rated power to be accomplished automatically by recirculation flow control alone, thus providing automatic electrical load-following capability for the ABWR without the need to adjust control rod settings.

The ABWR fine-motion control rod drives (FMCRDs) are moved electronically in small increments during normal operation, allowing precise power management. The FMCRDs are inserted into the core hydraulically during emergency shutdown, with the backup provision for continuous electronic insertion.

Description of safety concept

3.1 Safety requirements and design philosophy

Recognizing the need for continued safety enhancements in plant operation, one goal in designing the ABWR was to reduce core damage frequency by at least an order of magnitude relative to currently operating BWR plants. Essential design features contributing to this are enhancement of the high-pressure ECCS and RHR functions, including the emergency AC power supply, and the installation of diversified ATWS countermeasures. Furthermore, the adoption of reactor internal pumps (RIPs) eliminates large attached recirculation piping, particularly involving penetrations below the top of the core, and make it possible for a smaller emergency core cooling system (ECCS) network to maintain core coverage during a postulated loss of coolant accident.

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

The ABWR ECCS network was changed to a full three-division system, with both a high and low-pressure injection pump and heat removal capability in each division. For diversity, one of the systems, the RCIC system (a safety-grade system in the ABWR), includes a steam driven high-pressure pump. Transient response was improved by having three high-pressure injection systems available in addition to feedwater. The adoption of three on-site emergency diesel-generators to support core cooling and heat removal, as well as the addition of an on-site gas turbine-generator reduces the likelihood of "station blackout." The balanced ECCS system has less reliance on the automatic depressurization system (ADS) function, since a single motor-driven high-pressure core flooder (HPFL) is designed to maintain core coverage for any postulated line break size.

Response to anticipated transients without scram (ATWS) was improved by the adoption of the fine motion control rod drives (FMCRD), which allow reactor shutdown either by hydraulic or electric insertion. In addition, the need for rapid operator action to mitigate an ATWS was avoided by automation of emergency procedures, such as feedwater runback and standby liquid control (SLCS) injection.

3.3 Severe accidents (beyond design basis accidents)

The US ABWR also improved the capability to mitigate severe accidents even though such events are extremely unlikely. Through inerting, containment integrity threats from hydrogen generation were eliminated. Sufficient spreading area in the lower drywell, together with a passive drywell flooding system, assures coolability of postulated core debris. Manual connections make it possible to use on-site or off-site fire water systems to maintain core cooling. Finally, to reduce off-site consequences, a passive hard-piped wetwell vent, controlled by rupture disks set at twice design pressure (service level C), is designed to prevent catastrophic containment failure and provide maximum fission product “scrubbing.”

The result of this design effort is that in the event of a severe accident less than 0.25 Sv (25 rem) of radiation is released at the site boundary, even at a very low probability level. This means that the public's safety and health are assured. Figure 9.1 illustrates some of the severe accident mitigation features of 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

6.1.1 The main turbine

The main turbine is a six flow, tandem compound, single reheat, 1800 rpm machine with 1320.8 mm (52 in.) last stage blades. The turbine has one dual-exhaust high-pressure section and three dual-exhaust low-pressure sections. The cycle uses conventional moisture separator reheaters with single stage 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, respectively. The feedwater heating systems are designed to provide a final feedwater temperature of 216°C (420°F) at 100 percent nuclear boiling rate. This cycle yields a gross generator output of approximately 1420 MWe with a thermal reactor output of 3926 MW.

6.1.2 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 a three-valve chest that is connected to the main steam lines upstream of the turbine stop valves, and of three dump lines that separately connect each bypass valve outlet to one condenser shell. The system is designed to bypass at least 33% of the rated main steam flow directly to the condenser. The TBP, in combination with the reactor systems, provides the capability to shed 40% of the turbine generator rated load without reactor trip and without the operation of safety/relief valves. A load rejection in excess of 40% is expected to result in reactor trip but without operation of any steam safety valve.

The turbine bypass valves are opened by redundant signals received from the steam bypass and pressure control system whenever the actual steam pressure exceeds the preset steam pressure by a small margin. This occurs when the amount of steam generated by the reactor cannot be entirely used by the turbine. This bypass demand signal causes fluid pressure to be applied to the operating cylinder, which opens the first of the individual valves. As the bypass demand increases, additional bypass valves are opened, dumping the steam to the condenser. The bypass valves are equipped with fast acting servo valves to allow rapid opening of bypass valves upon turbine trip or generator load rejection.

6.1.3 Main condenser

The main condenser, which does not serve or support any safety function and has no safety design basis, is a multi-pressure 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 in series through the three shells. The condenser circulating water system (CCW) is designed to permit any portion of the condenser to be isolated and removed from service.

The main condenser is located in the turbine building in pits below the operating floor and is supported by the turbine building basemat. The condensate return tank is located in the turbine building above its connection to the low-pressure condenser shell.

Since the main condenser operates at a vacuum, radioactive leakage to the atmosphere cannot occur. Circulating water leakage into the shell side of the main condenser is detected by measuring the conductivity of the condensate. Conductivity of the condensate is continuously monitored at selected locations in the condenser. Leak detection trays are included at all tube-to-tubesheet interfaces. Provisions for early leak detection are provided at tubesheet trays and in each hotwell section. The hotwell is divided into sections to allow for leak detection and location. Conductivity and sodium content are alarmed in the main control room and preclude any automatic bypass of the demineralizers.

The main condenser evacuation system (MCES) removes the non-condensable gases from the power cycle. The MCES removes the hydrogen and oxygen produced by radiolysis of water in the reactor, and other power cycle non-condensable gases, and exhausts them to the offgas system during plant power operation, and to the turbine building compartment exhaust system at the beginning of each startup.

The MCES consists of two 100%-capacity, double stage, steam jet air ejector (SJAE) units (complete with intercondenser) for power plant operation where one SJAE unit is normally in operation and the other is on standby, as well as a mechanical vacuum pump for use during startup. The last stage of the SJAE is a non-condensing stage.

During the initial phase of startup, when the desired rate of air and gas removal exceeds the capacity of the steam jet air ejectors, and nuclear steam pressure is not adequate to operate the SJAE units, the mechanical vacuum pump establishes a vacuum in the main condenser and other parts of the power cycle. The discharge from the vacuum pump is then routed to the turbine building compartment exhaust system, since there is then little or no effluent radioactivity present. Radiation detectors in the turbine building compartment exhaust system and plant vent alarm in the main control room if abnormal radioactivity is detected. Radiation monitors are provided on the main steam lines which trip the vacuum pump if abnormal radioactivity is detected in the steam being supplied to the condenser.

The SJAEs are placed in service to remove the gases from the main condenser after a pressure of about 34 to 51 kPa absolute is established in the main condenser by the mechanical vacuum pump and when sufficient nuclear steam pressure is available.

During normal power operations, the SJAEs are normally driven by cross-around steam, with the main steam supply on automatic standby. The main steam supply, however, is normally used during startup and low load operation, and auxiliary steam is available for normal use of the SJAEs during early startup, should the mechanical vacuum pump prove to be unavailable.

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 off-gas condenser, the condensate demineralizer, and through three parallel strings of four low pressure feedwater heaters to the reactor feed pump section. The two reactor feed pumps each have an approximate capacity of 4600 m^3/h. They each discharge through two stages of high-pressure heaters (two parallel strings) to the reactor. Each reactor feedwater pump is driven by an adjustable speed synchronous motor. The drains from the high-pressure heaters are pumped backward to the suction of the feed pumps.

Two 22 in. (559 mm) feedwater lines transport feedwater from the feedwater pipes in the steam tunnel through RCCV penetrations to horizontal headers in the upper drywell that have three 12 in. (305 mm) riser lines connecting to nozzles on the RPV. Isolation check valves are installed upstream and downstream of the RCCV penetrations and manual maintenance gate valves are installed in the 22-in. lines upstream of the horizontal headers.

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 (TBSW) system. The TBSW system rejects the heat taken from the TBCW system to the power cycle heat sink that is part of the circulating water system.

The service air (SAIR) system provides compressed air for general plant use. The SAIR system also provides backup to the instrument air (IAIR) system in the event that the IAIR system pressure is lost. The IAIR system provides compressed air for pneumatic equipment, valves, controls and instrumentation outside the primary containment.

7.1 Operational power supply systems

On-site power is supplied from the plant turbine generator, utility power grid or an off-site power source, depending on the plant operating status. During normal operation, plant loads are supplied from the main generator through the unit auxiliary transformers. A generator breaker allows the unit auxiliary transformers to stay connected to the grid to supply loads by backfeeding from the switchyard when the turbine is not online.

Direct current power supply

The DC power supply system (DC) consists of three separate subsystems:

- safety related 125 V DC,
- non-safety related 250 V DC, and
- non-safety related 125 V DC.

The system begins at the source terminals of the plant safety and non-safety battery chargers. It ends at the input terminals of the plant DC loads (motor, control loads, etc.) and at the input terminals of the inverters of the low voltage vital AC power supply system.

Each DC subsystem consists of a battery, associated battery charger, power distribution panels, and all the associated control, monitoring and protective equipment and interconnecting cabling. In addition, DC employs standby chargers that are shared between the batteries to enable the individual battery testing and off-line equalization.

DC operates with its battery and battery chargers (except standby chargers) continuously connected to the DC system. During normal operation, the DC loads are powered from the battery chargers with the batteries receiving a continuous charging current (i.e., floating) on the system. In case of loss of AC power to the charger or its failure, the DC loads are automatically powered from the batteries.

Instrument and control power supply

The instrument and control power supply system (ICP) provides 120 V AC power to instrument and control loads that do not require continuity of power during a loss of preferred power.

The ICP system consists of class 1E and non-Class 1E interruptible power supplies and their respective regulating step-down power transformers (conditioners), a transfer switch (for non-class 1E subsystem only), alternating current (AC) distribution panels, and cables to the distribution system loads.

The ICP system is powered from 480V motor control centers (MCC) and is distributed at 208Y/120V. Power conditioners are used as voltage regulating transformers to regulate its output voltage to various I&C loads under broad variations in supply voltage and load changes. Power conditioners are sized to supply their respective I&C loads under the most demanding operating conditions.

7.2 Safety-related electrical systems

7.2.1 Class IE AC power supply

The class 1E buses of the on-site power system consist of three independent divisions of class 1E equipment. Each division is fed by an independent class 1E bus at the medium voltage level, and each division has access to one on-site and two off-site (normal and alternate preferred) power sources. Each division has access to an additional power source, which is provided by the combustion turbine generator (CTG).

Each division is provided with an on-site safety related standby diesel generator that supplies a separate on-site source of power for its division when normal or alternate preferred power is not available. The standby diesel generators are capable of providing the required power to safely shut down the reactor after loss of preferred power and/or loss of coolant accident and to maintain the safe shutdown condition and operate the class 1E auxiliaries necessary for plant safety after shutdown.

The on-site standby AC power supplies (diesel generators) have sufficient capacity to provide power to all their respective loads. Loss of the preferred power supply, as detected by undervoltage relays in each division, will cause the standby power supplies to start and automatically connect, in sufficient time to safely shut down the reactor or limit the consequences of a design basis accident (DBA) to acceptable limits, and maintain the reactor in a safe condition.

7.2.2 Direct current power supply

The class 1E 125 V DC subsystem consists of four independent and redundant divisions (I, II, III, and IV). All four divisional batteries are sized to supply 125 V DC power to their loads during a design basis accident, coincident with loss of AC power, for a period of at least two hours based on the most limiting load profile without load shedding. This sizing of the division I battery also meets the requirement to permit operation of the station blackout coping systems for eight hours with manual load shedding. This manual load shedding commences only after the first two hours of station blackout and includes the vital AC power, as well as the remote multiplexing units (RMU) and division I diesel generator control loads. The division I battery is sized to support operation of the reactor core isolation cooling (RCIC) system and remote shutdown system (RSD), as well as minimum necessary emergency lighting. This manual load shedding takes credit for the RCIC operation from outside the main control room.

7.2.3 Vital (uninterruptable) power supply

The class 1E vital AC (VAC) power supply provides redundant, reliable power to the safety logic and control functions during normal, upset and accident conditions. The VAC is comprised of three independent subsystems. Each subsystem supplies uninterruptible, regulated AC power to those loads that require continuity of power during a loss of preferred power (LOPP).

Each VAC subsystem's division or load group is comprised of an independent uninterruptible power supply, maintenance bypass switch, regulating transformers, main distribution panel, local distribution panels, and cables for power, instrumentation and control. Each uninterruptible power supply is a constant voltage constant frequency (CVCF) inverter power supply unit consisting of a rectifier, inverter, and AC and DC static transfer switches. Each CVCF power supply is provided with an alternate AC source with sufficient capacity to allow normal operation in case of failure or unavailability of a single inverter.

7.3 I&C design concepts, including control room

The ABWR control and instrument systems are designed to provide manual and automatic means to control plant operations and initiate protective actions should plant upset conditions occur. The ABWR utilizes digital controllers, interfacing with plant equipment, sensors and operator controls through a multiplexing system for signal transmission to achieve these functions.

The key distinguishing simplification features for plant control and monitoring include:

- Enhanced man-machine interface design
- Automated plant operations
- Simplified neutron monitoring system
- Reduction in number of nuclear boiler instruments
- Fault-tolerant safety system logic and control
- Standardized digital control and measurement
- Multiplexing of plant control signals.

Multiplexed signal transmission using high-speed fiber optic data links is combined with digital technology to integrate control and data acquisition for both reactor and turbine plants. Multiplexing significantly reduces the quantities of control cables that need to be installed during construction, thereby reducing the construction cost, and facilitates automation of plant operations.

Performance monitoring and control, and power generator control subsystem functions are provided by the process computer system to support efficient plant operation and automation.

The main control room panels (MCRPs) consist of an integrated set of operator interface panels (e.g., main control console, large display panel), as depicted in Figure 7.1. The safety related panels are seismically qualified and provide grounding, electrical independence and physical separation between safety divisions and non-safety-related components and wiring.

The MCRPs and other main control room operator interfaces are designed to provide the operator with information and controls needed to safely operate the plant in all modes, including startup, refuelling, safe shutdown, and maintaining the plant in a safe shutdown condition. Human factors engineering principles have been incorporated into all aspects of the ABWR main control room design.

The liquid and solid radwaste systems are operated from control panels in the radwaste control room. Programmable controllers are used in this application.

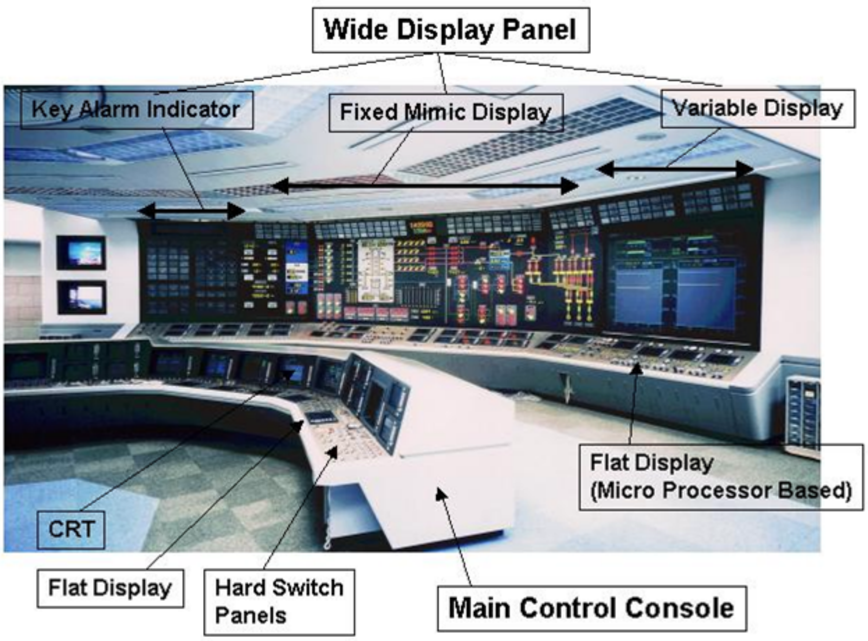

FIG. 7.1. ABWR – main control room (KK-6)

7.4 Reactor protection and other I&C safety systems

The safety system logic and control (SSLC) provides a centralized facility of implementing safety related logic functions. The SSLC is configured as a four-division data acquisition and control system, with each division containing an independent set of microprocessor-based software controlled logic processors.

The reactor protection system (RPS) is an overall complex of instrument channels, trip logic, trip actuators, manual controls, and scram logic circuitry that initiates the rapid insertion of control rods by hydraulic force to scram the reactor when unsafe conditions are detected. The RPS uses the functions of the essential multiplexing subsystem (EMS) and the SSLC system to perform its functions.

The remote shutdown system (RSD) is designed to safety shut down the reactor from outside the main control room. The RSD provides remote manual control to the systems necessary to: (a) achieve prompt hot shutdown of the reactor after a scram; (b) achieve subsequent cold shutdown of the reactor; and, (c) maintain safe conditions during shutdown.

The standby liquid control (SLC) system is designed to provide an alternate method of reactor shutdown from full power to cold subcritical by the injection of a neutron absorbing solution to the RPV.

The feedwater control (FWC) system controls the flow of feedwater into the RPV to maintain the water level in the vessel within predetermined limits during all plant operating modes.

The neutron monitoring system (NMS) is a system of in-core neutron detectors and out-of-core electronic monitoring equipment. The system is designed to provide indication of neutron flux, which can be correlated to thermal power level for the entire range of flux conditions that can exist in the core. There are four subsystems in the NMS: the startup range neutron monitoring (SRNM) subsystem; the power range neutron monitoring (PRNM) subsystem [comprised of the local power range monitors (LPRM) and average power range monitors (APRM)]; the automatic traversing in-core probe (ATIP) subsystem; and, the multi-channel rod block monitoring (MRBM) subsystem.

7.4.1 Startup range neutron monitoring (SRNM) subsystem

The SRNM subsystem monitors the neutron flux from the source range to 15% of the rated power. The SRNM is designed to provide neutron flux related trip inputs (flux level and period) to the RPS, including a non-coincident trip function for refuelling operations and a coincident trip function for other modes of operation. The SRNM has 10 channels where each channel includes one detector installed at a fixed position within the core.

7.4.2 Power range neutron monitoring (PRNM) subsystem

The PRNM subsystem provides flux information for monitoring of the average power level of the reactor core. It also provides information for monitoring of the local power level. The PRNM is used when the reactor power is above approximately 1% of rated power.

The PRNM subsystem consists of two subsystems:

- Local power range monitoring (LPRM) subsystem,
- Average power range monitoring (APRM) subsystem.

The LPRM subsystem continuously monitors local core neutron flux. It consists of 52 detector assemblies with 4 detectors per assembly. The 208 LPRM detectors are separated and divided into four groups to provide four independent APRM signals. The APRM subsystem averages the readings of the assigned LPRM detectors and provides measurement of reactor core power. Individual LPRM signals are also transmitted through dedicated interface units to various systems such as the reactor control and instrumentation system (RC&IS), and the plant process computer.

7.4.3 Automatic traversing in-core probe (ATIP) subsystem

The ATIP subsystem performs an axial scan of the neutron flux in the core at the LPRM assembly locations. The subsystem can be controlled manually by the operator, or it can be under micro-processor-based automated control. The ATIP subsystem consists of neutron-sensitive ion chambers, flexible drive cables, guide tubes, indexing machines, drive machines, and an automatic control system. Working in conjunction with the performance monitoring and control system (PMCS), the ATIP subsystem calibrates the LPRM outputs.

7.4.4 Multi-channel rod block monitor (MRBM) subsystem

The MRBM subsystem is designed to stop the withdrawal of control rods and prevent fuel damage when the rods are incorrectly being continuously withdrawn, whether due to malfunction or operator error. The MRBM averages the LPRM signals surrounding each control rod being withdrawn. It compares the averaged LPRM signal to a preset rod block setpoint, and, if the averaged value exceeds this setpoint, the MRBM subsystem issues a control rod block demand to the RC&IS. The rod block setpoint is a core flow biased variable setpoint.

Spent fuel and waste management

No information provided.

Plant layout

9.1 Buildings and structures, including plot plan

The ABWR plant includes all buildings that are dedicated exclusively or primarily to housing systems and the equipment related to the nuclear system, or that control access to these systems. There are five such buildings within the scope:

a. Reactor building - includes the reactor pressure vessel, containment, and major portions of the nuclear steam supply system, refuelling area, diesel generators, essential power, non-essential power, emergency core cooling systems, heating, ventilation and cooling (HVAC) system and supporting systems.
b. Service building - personnel facilities, security offices, and health physics station.
c. Control building - includes the control room, the computer facility, reactor building component cooling water system and the control room HVAC system.
d. Turbine building - houses all equipment associated with the main turbine generator. Other auxiliary equipment is also located in this building.
e. Radwaste building - houses all equipment associated with the collection and processing of solid and liquid radioactive waste generated by the plant.
f. The site plan of the ABWR includes the reactor, service, control, turbine, radwaste and supporting buildings. Provision is made within the reactor building for 10 years spent fuel storage. Separate buildings can be provided for additional on-site waste storage and spent fuel storage for 20 years. Figure 9.2 illustrates the site plan of the ABWR.

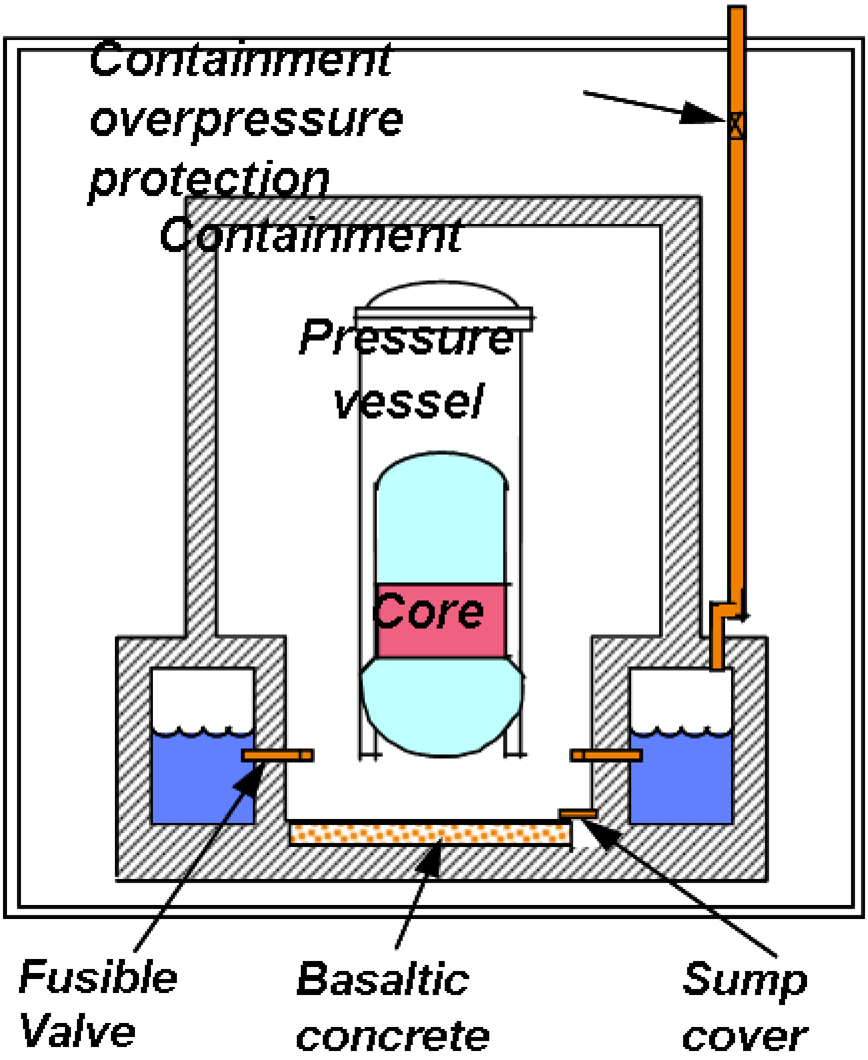

FIG. 9.1. ABWR - severe accident mitigation features

Development of the ABWR plant and building arrangements has been guided by the following criteria:

a. Retain the passive and well-established BWR pressure suppression containment technology. Use of the horizontal vent configuration confirmed for the Mark III containments.
b. Emphasize optimal layout of systems to improve personnel access and equipment maintenance activities.
c. Locate major equipment for early installation using open top construction approach and large-scale modularization.
d. Arrange the reactor building around the primary containment to provide multiple barriers to post-accident fission product leakage, and high tolerance to external missiles.

The ABWR design arrangement minimizes material quantities. This, when combined with the volume reduction, contributes to the substantial reduction in both the construction schedule and plant capital cost.

The layout of the reactor and turbine buildings was based on the following considerations:

a. Personnel access for all normal operating and maintenance activities was a primary concern starting with the first layout studies. Access routes from the change room to contaminated reactor and turbine building areas are as direct as possible. At each floor, 360° access is provided, if practical, to enhance daily inspections and normal work activities. Access to equipment not reachable from floor level is via platform and stair access wherever possible.
b. Equipment access is provided for all surveillance, maintenance and replacement activities with local service

areas and laydown space for periodic inspections. Adequate hallways and other equipment removal paths, including vertical access hatches, are provided for moving equipment from its installed position to service areas or out of the building for repair. Lifting points, monorails and other installed devices are provided to facilitate equipment handling and minimize the need for re-rigging individual equipment movements. The equipment access also considers the need for construction access.

c. Radiation levels for personnel are controlled and minimized. The reactor building is divided into clean and controlled areas. Once personnel enter a clean or controlled area, it is not possible to crossover to the other area without returning to the change area. Redundant equipment is located in shielded cells to permit servicing one piece of equipment while the plant continues to operate. Valve galleries are provided to minimize personnel exposure during system operation or preparation for maintenance.

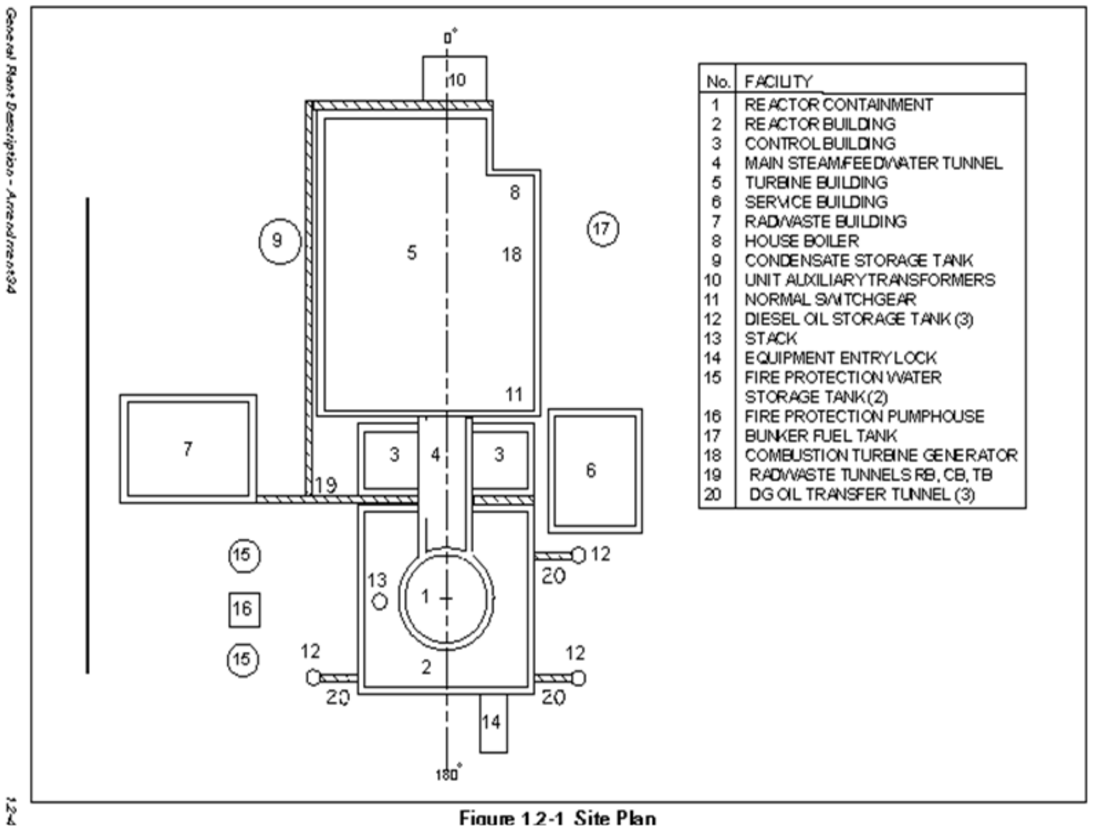

FIG. 9.2. ABWR - site plan

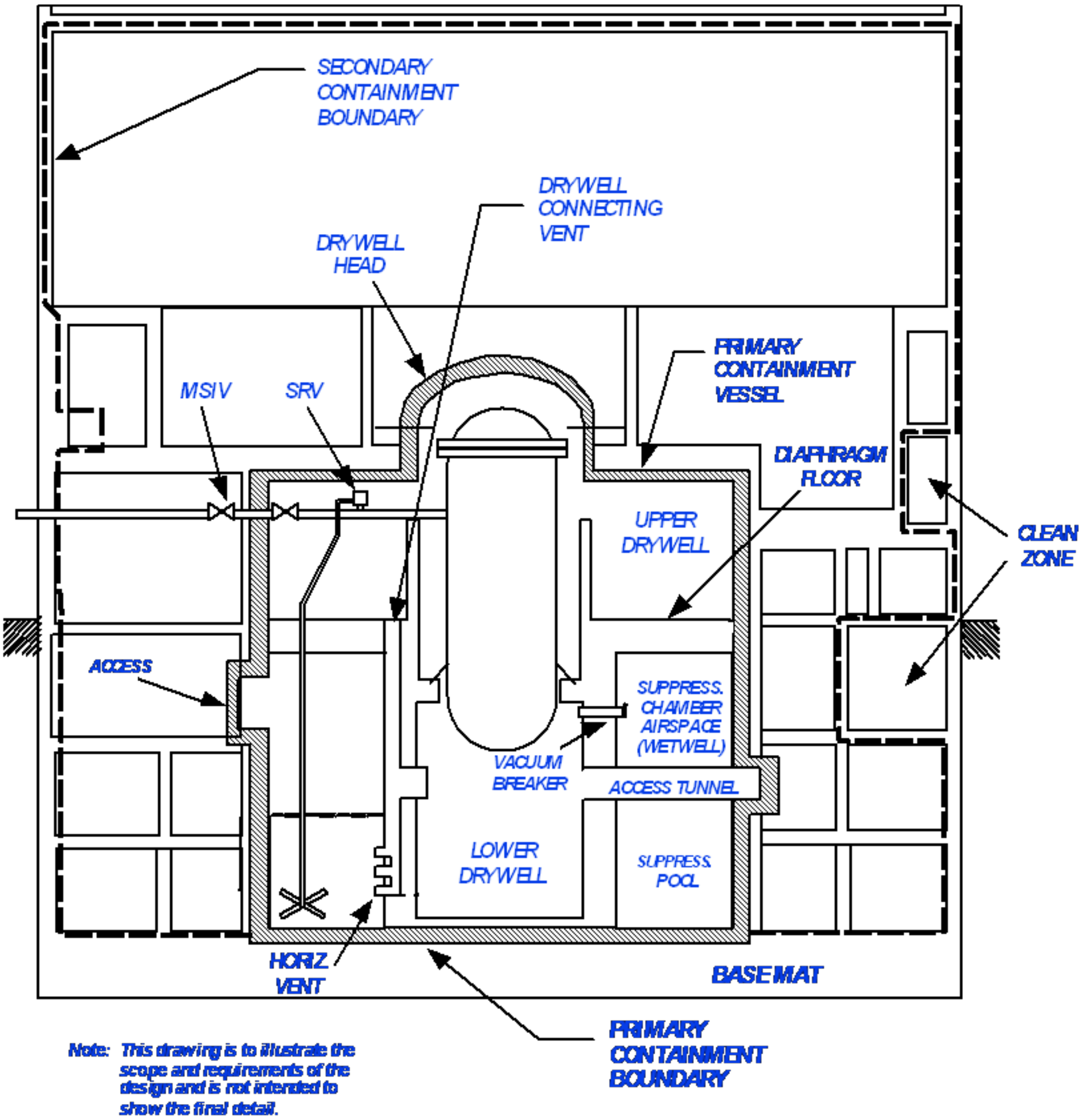

FIG. 9.3. ABWR - Containment structure features

The turbine generator is aligned with its axis in-line with the reactor building. This is done to minimize the possibility of turbine missile impact on the containment vessel.

The main and auxiliary transformers are located adjacent to the main generator at the end of the turbine building. This location minimizes the length of the isophase bus duct between the generator and transformers, as well as the power supply cables back to the main electrical area of the power block.

The site plan includes consideration for construction access. The arrangement provides a clear access space around the reactor and turbine buildings for heavy lift mobile construction cranes without interference with other cranes, access ways and miscellaneous equipment.

9.2 Reactor building

The ABWR reactor building is a reinforced concrete structure. The integrated reactor building and containment structure has been analyzed for a safe shutdown earthquake of 0.3g.

A secondary containment surrounds the primary containment and provides a second containment function including a standby gas treatment system. Off-site radiological dose studies have shown that a containment leak rate of less than 0.5%/day is achievable.

Careful attention has been given to ease of construction with this building arrangement. The building features full 360° access on all floors for ease of worker movement. Generally, the major cooling equipment has been placed on the lowest floors of the building to allow early installation during construction.

Modularization techniques are implemented to reduce costs and improve construction schedules. These techniques are applied to such reactor building items as (1) building reinforcing bar assemblies, (2) structural steel assemblies, (3) steel liners for the containment and associated water pools, and (4) selected equipment assemblies.

Removal of the post LOCA decay heat is achieved by the containment heat removal system, consisting of the suppression pool cooling mode, wetwell, and drywell and drywell spray features. An integral part of the RHR system, the system removes steam directly from the drywell and wetwell into the suppression pool. The large volume of water in the suppression pool serves as a fission product scrubbing and retention mechanism. The reactor building serves as an additional barrier between the primary containment and the environment. Any fission product leakage from the primary containment is confined within the reactor building.

Analyses of the radiological dose consequences for accidents, based on an assumed containment leak rate of 0.5% per day, show that the off-site dose after an accident is less than 1 rem. This favourable dose rate is made possible by trapping fission products within the secondary containment with a slight negative pressure and processing the air through the standby gas treatment system.

Key distinguishing features of the ABWR reactor building design include:

a. Elimination of external recirculation loops reduces the containment volume associated with high construction costs.
b. Reduced building volume reduces material costs and construction schedule.
c. Design with simple structural shapes to improve constructability to reduce capital costs and the construction schedule.
d. Improved personnel and equipment access for enhanced operability and maintainability.

The volume of the ABWR reactor building has been reduced to approximately 167 000 cubic meters. Since this reduced volume was obtained by simplification of the reactor supporting systems and optimization of their arrangement with improved access (rather than simply by compaction), it provides material cost savings and helps reduce the construction schedule without adversely impacting maintenance.

The major equipment access to the reactor building is via double door vestibule at grade level. This entry area is connected to the refuelling floor by a large hatch serviced by the reactor building crane. The reactor building layout utilizes the grade level entry area for major servicing of the cooling equipment. All of the major pieces of equipment can be moved into the area through hatches.

9.3 Containment

The ABWR pressure suppression primary containment system comprises the drywell (DW), wetwell, and supporting systems. The main features of the ABWR containment structure are illustrated in Figure 9.3. A reinforced concrete containment vessel (RCCV) with an internal steel liner was adopted as the primary containment vessel (PCV) of ABWR. It is united with the reactor building (R/B) structure except for a drywell head and other penetrations. The steel RPV pedestal and the reinforced concrete diaphragm floor, partition the containment volume into a drywell and suppression chamber. The drywell and suppression chamber are connected by the steel vent pipes installed between a double shell steel structure of the RPV pedestal.

9.4 Turbine building

The turbine building houses all the components of the power conversion system. This includes the turbine-generator, main condenser, air ejector, steam packing exhauster, off-gas condenser, main steam system, turbine bypass system, condensate demineralizers, and the condensate and feedwater pumping and heating equipment. The small size of the ABWR turbine building makes a significant contribution to capital cost savings and a shorter construction schedule.

As aforementioned, the ABWR was designed to improve safety, operation and maintenance (O&M) practices, economics, radiation exposure and so on. This section points out which corresponding technology results in what kind of benefit category.

10.1 Design simplification

The direct cycle system of BWR is originally more simplified than dual cycle system of PWR. Furthermore, ABWR substitutes RIP (refer to 2.4 Primary components) for large primary loop recirculation pumps and piping of conventional BWR. This is an evolutionary simplified design that can condense the nuclear boiler system only within RPV attached no recirculation piping.

10.2 Operation flexibility improvement

Typical examples are FMCRD and the new-designed main control room to enhance ABWR operational flexibility. The FMCRD is described in 2.2 Reactor core and fuel design, and advanced main control room is in 7.3. I&C design concepts, including control room

10.3 Cost reduction of equipment and structures

Equipment reduction (typical example):

As mentioned in 3.2 Safety systems and features (active, passive and inherent), a full three-division system, with both a high and low pressure injection pump and heat removal capability in each division, is adopted, however capacity of ECCS is rather much reduced.

Structure reduction:

The Reactor Building volume is reduced to 167000 cubic meters, led by optimized equipment and piping arrangement and adoption of RCCV, providing material cost savings. (refer 9.2 reactor building).

10.4 Reduction of construction period

Modularization techniques are the most effective for the short construction period of the ABWR (refer 9.2 reactor building) in addition to the building volume reduction aforementioned.

10.5 Scope reduction of the maintenance during operation and outages

The typical example of maintenance reduction is:

There are 103 HCUs for 205 FMCRDs, each HCU provides sufficient volume of water stored at high pressure in a pre-charged accumulator to scram two FMCRDs at any reactor pressure.

10.6 Making the maintenance easier and with lower radiation exposure

The typical example of easy maintenance:

The sealless FMCRD 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 with lower radiation exposure. (Refer 2.2 Reactor core and fuel design)

Lower radiation exposure:

Reduction in radioactive nuclide concentration in reactor water, control of radioactive nuclide deposition and optimization of the permanent radiation shielding result in radiation level reduction.

10.7 Increasing the capacity factor

Such total improvement results in actual ABWRs' excellent operating experiences. (Refer IAEA-TECDOC-1245, Ref. [2]).

10.8 Reduction of the power generating cost

The large, low-pressure turbine with a 52-inch last stage blade (LSB), the moisture separator/heater (MSH), and the heater drain pump system, are adopted to increase the turbine system efficiency. (Refer 6.1 Turbine generator plant) Furthermore, the drums of radioactive waste discharged are reduced by adoption of latest radioactive waste management system. (Refer IAEA-TECDOC-1175, Ref. [3]).

Development status of technologies relevant to the NPP

No information provided.

Deployment status and planned schedule

This section provides latest project status after the former report "IAEA-TECDOC-968"[1]. The status before 1997 is described in "IAEA-TECDOC-968".

12.1 Entities involved (Japan)

The Japanese version of the ABWR was developed by GE, Hitachi Ltd. and Toshiba Corp. under the sponsorship of the Tokyo Electric Power Company (TEPCO). In 1987, TEPCO announced its decision to proceed with a two-unit ABWR project at its Kashiwazaki-Kariwa nuclear power station, 220 kilometers northwest of Tokyo, as Kashiwazaki-Kariwa nuclear Power Station Unit 6&7. KK-6 and –7 began commercial operation in November 1996 and July 1997, respectively. Each is rated 1,315MWe (net). The results of the first ten reactor years of combined operating experience for TEPCO KK-6&7 indicated below:

- The ABWRs are performing up to expectations
- Unplanned shutdowns have been due to conventional problems and do not suggest there are any ABWR-specific problems
- BWR technology is becoming safer and more economic
- Compared to earlier BWR technology, ABWRs have lower occupational radiation exposure, increased availability, higher load factors and lower O&M costs
- ABWRs would operate more efficiently under less severe operating constraints and with improved management strategies

Ten ABWR deployment programmes are underway in Japan as follows:

* Hamaoka-5, commenced commercial operation 2005

(Chubu Electric Power Co. Inc.)

* Shika-2, commenced commercial operation 2006

(Hokuriku Electric Power Co.)

* Shimane-3 c/o 2011 under construction (Chugoku Electric Power Co.)

* Fukushima-I, 7&8 c/o 2015 & 2016 EIS submitted (TEPCO)

* Ohma (Full-MOX) c/o 2014 under construction (EPDC)

* Higashidori-1&2 c/o 2017 & 2019 Site Authorized (TEPCO)

* Kaminoseki 1&2 c/o 2015 & 2020 Site Authorized (Chugoku Electric Power Co.)

FIG. 12.1. Kashiwazaki-Kariwa Nuclear Power Station Unit 6&7 (Tokyo Electric Power Co.) (Right Photo: from left, Unit 7, Unit 6 (ABWR, 1356MWe) and Unit 5 (BWR-5, 1100MWe))

12.2 Entities involved (US)

First of a kind engineering (FOAKE) has been conducted for the ABWR design for application in the United States. Funding for the ABWR FOAKE project was provided by GE and its FOAKE associates including members of the ABWR FOAKE design team, the Advanced Reactor Corporation (ARC), representing utility sponsors of the ABWR FOAKE project, and the United States Department of Energy (DOE). The ABWR FOAKE Project began in June 1993 and was completed in 1996.

The result of FOAKE and Design Certificate from U.S.NRC makes it possible to construct ABWRs in the US from the viewpoint of preliminary engineering and reactor type certified. In fact, it was also reflected in the Taiwan, China project as below.

12.3 Entities involved (Taiwan, China)

Through a competitive bidding process, Taiwan Power Co. (TPC) selected the ABWR for its two unit Lungmen project. GE designed and provided the scope of supply for two 1,350 MWe ABWRs. The Lungmen project is supported by the GE Team, including: Black & Veatch, Hitachi, Shimizu, Toshiba, and other US, Taiwan, China and international participants.

12.4 Design status and Licensing process in US

On September 29, 1987, GE applied for certification of the U.S.ABWR standard design with the US Nuclear Regulatory Commission (U.S.NRC). The U.S.NRC staff issued a final safety evaluation report (FSER) related to the certification of the U.S.ABWR design in July 1994 (NUREG-1503). The FSER documents the results of the NRC staff's safety review of the U.S.ABWR design against the requirements of 10 CFR Part 52, Subpart B, and delineates the scope of the technical details considered in evaluating the proposed design. Subsequently, the applicant submitted changes to the U.S.ABWR design and the NRC staff evaluated these design changes in a supplement to the FSER (NUREG-1503, Supplement No.1).

U.S.NRC adopted as final this design certification rule, Appendix A to 10 CFR Part 52, for the U.S. ABWR design in May 1997.

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Technical data

General plant data

Reactor thermal output	3926 MWth
Power plant output, gross	1420 MWe
Power plant output, net	1350 MWe
Power plant efficiency, net	34.4 %
Plant design life	60 Years
Plant availability target >	87 %
Primary coolant material	Light Water
Moderator material	Light water
Thermodynamic cycle	Rankine
Type of cycle	Direct

Safety goals

Core damage frequency <	10E-5 /Reactor-Year
Large early release frequency <	10E-6 /Reactor-Year
Occupational radiation exposure <	1.0 Person-Sv/RY

Nuclear steam supply system

Steam flow rate at nominal conditions	2122 Kg/s
Steam pressure	7.07 MPa(a)
Steam temperature	287.8 °C
Feedwater flow rate at nominal conditions	2118 Kg/s
Feedwater temperature	215.6 °C

Reactor coolant system

Primary coolant flow rate	14502 Kg/s
Reactor operating pressure	7.07 MPa(a)
Core coolant inlet temperature	278 °C
Core coolant outlet temperature	288 °C
Mean temperature rise across core	10 °C

Reactor core

Active core height	3.81 m
Equivalent core diameter	5.163 m
Average linear heat rate	13.3 KW/m
Average fuel power density	25.0 KW/KgU
Average core power density	49.2 MW/m^3
Fuel material	Sintered UO2
Outer diameter of fuel rods	10.3 mm
Rod array of a fuel assembly	10x10
Lattice geometry	Square
Number of fuel assemblies	872
Enrichment of reload fuel at equilibrium core	4 Weight %
Fuel cycle length	24 Months

Average discharge burnup of fuel	50 MWd/Kg
Burnable absorber (strategy/material)	Gd2O3
Control rod absorber material	B4C and Hf
Soluble neutron absorber	Boron

Reactor pressure vessel

Inner diameter of cylindrical shell	7100 mm
Wall thickness of cylindrical shell	190 mm
Design pressure	8.62 MPa(a)
Design temperature	301.7 °C
Base material	SA508, CLASS 3
Total height, inside	21000 mm
Transport weight	1264 t

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	28.7 m
Flow at rated conditions	14.53 m^3/s

Primary containment

Type	Pressure-suppresion reinforced concrete
Overall form (spherical/cylindrical)	Cylindrical
Dimensions - diameter	29 m
Dimensions - height	36.1 m
Design pressure	0.310 MPa
Design temperature	171.1 °C
Design leakage rate	0.5 Volume % /day

Turbine

Type of turbines	Six flow, Tandem compound, Single reheat

Number of turbine sections per unit (e.g. HP/MP/LP)	1/0/3
Turbine speed	1800 rpm
HP turbine inlet pressure	6.792 MPa(a)
HP turbine inlet temperature	283.7 °C

Generator

Type	Three-phase, turbo generator
Rated power	1580 MVA
Active power	1420 MW
Voltage	27 kV
Frequency	60 Hz
Total generator mass including exciter	730 t

Condenser

Type	Shell type
Condenser pressure	11.75 kPa

Feedwater pumps

Number	3
Pump speed	5000 rpm
Head at rated conditions	60 m
Flow at rated conditions	1.0 m^3/s