Status report 80 - Reduced-Moderation Water Reactor (RMWR)

Overview

Full name	Reduced-Moderation Water Reactor
Acronym	RMWR
Reactor type	Boiling Water reactor (BWR)
Coolant	Light Water
Moderator	Light water
Neutron spectrum	Thermal Neutrons
Thermal capacity	3926.00 MWth
Electrical capacity	1356.00 MWe
Design status	Conceptual Design
Designers	JAEA
Last update	04-04-2011

Description

Introduction

The purposes of the Reduced-Moderation Water Reactor (RMWR) are to ensure the sustainable energy supply, to meet the flexible demand of plutonium, and to reduce the spent fuel accumulation, being based on the well-experienced LWR technologies. In order to accomplish these purposes, a high conversion ratio beyond 1.0 is an essential requirement from the reactor physics point of view. To obtain such a high conversion ratio in a LWR core, the neutron energy should be increased by reducing the water to fuel volume ratio. The in-core water volume can be reduced by introducing tight-lattice fuel assemblies. Increasing the void fraction also contributes to the reduction of neutron moderation in the core. The resultant neutron spectrum is similar to that in fast breeder reactors (FBRs), and much harder than that in LWRs as shown in Figure 1.

An effective utilization of uranium resources can be attained by breeding the fissile materials as in the case of FBR. Figure 2 shows a comparison of cumulative consumption of natural uranium with and without introducing RMWR. In the analysis shown in Figure 2, nuclear power capacity in Japan is assumed to be constant at 80 GWe after year 2050. When RMWRs are introduced, natural uranium consumption can be limited to moderate finite levels because the conversion ratio exceeds one.

The high conversion ratio also results in maintaining the plutonium quality, which makes it possible to recycle plutonium many times, i.e. multiple recycling, as shown in Figure 3. This is a good method for deployment of plutonium and contributes to the reduction of spent fuel accumulation for the long term.

The excess reactivity for the high burn-up and long operation cycle is expected to be reduced due to the high conversion ratio of RMWR as shown in Figure 4. The characteristics of high burn-up and long operation cycle are very beneficial to reduce fuel cycle cost as well as reducing the periodical inspection cost.

The representative concept of RMWR was developed by Japan Atomic Energy Research Institute (JAERI), which is now Japan Atomic Energy Agency (JAEA), in collaboration with the Japan Atomic Power Company (JAPC) and Hitachi, Toshiba and Mitsubishi Heavy Industries. Apart from the RMWR concept of which the conversion ratio exceeds 1.0, another core concept which has a conversion ratio of 0.8~0.9 has also been developed by JAEA. A combination of RMWR and this core concept is named Innovative Water Reactor for Flexible Fuel Cycle (FLWR). Then the core concept is named High Conversion Type FLWR (HC-FLWR). The concept of HC-FLWR has been created putting priority on keeping smooth technical continuity with the current LWR and MOX-LWR technologies. Therefore, HC-FLWR has a more conventional core design than RMWR. For example, the gap width between fuel rods is wider than that of RMWR. The MOX fuel has a lower fissile plutonium content than in RMWR. The coolant void fraction is lower than that of RMWR. The discharge burn-up is lower than that of RMWR. HC-FLWR can be introduced easily as a replacement for the current LWR because of its more conventional design. As HC-FLWR uses a fuel assembly with the same external shape as that of RMWR, it can be converted to RMWR by replacing only the fuel assemblies without any change of reactor systems when the fuel cycle for plutonium multiple recycling with MOX fuel reprocessing is realized. HC-FLWR thus can be a bridge between the current LWR and RMWR.

The R&D activities include core and system design studies, thermal-hydraulic experiments, reactor physics experiments, and safety analyses. Although design studies were performed both for BWR-type and PWR-type cores, only the BWR-type system is described in the following sections, as a representative RMWR.

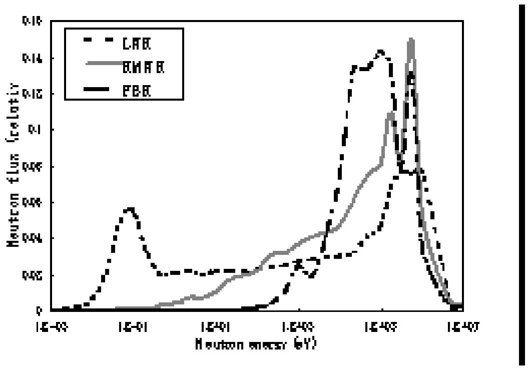

FIG. 1. Comparison of neutron spectrum among LWR, RMWR and FBR

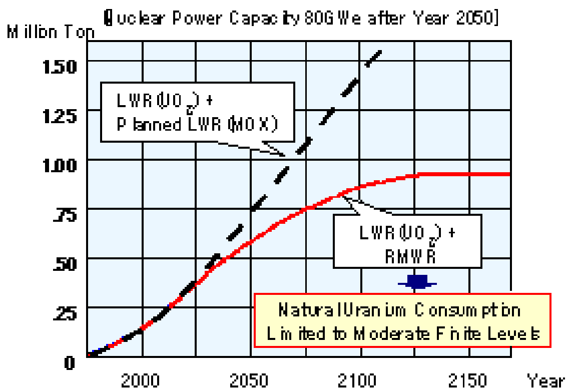

FIG. 2. Cumulative consumption of natural uranium

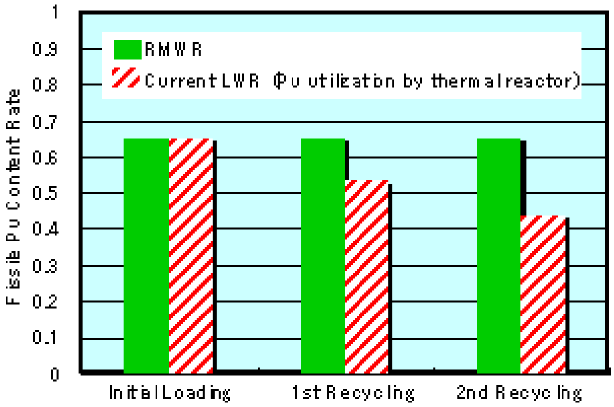

FIG. 3. Change of the fissile Pu content rate with the recycling

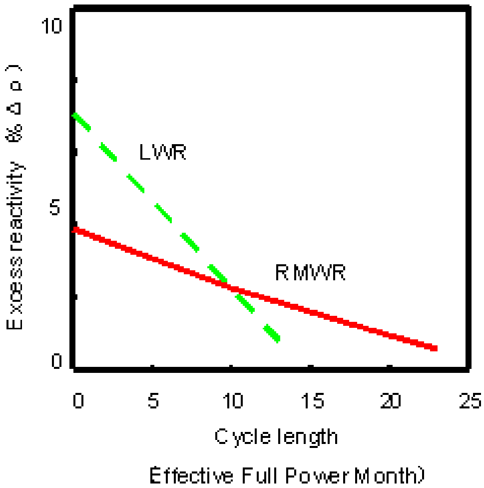

FIG. 4. Decrease in excess reactivity with burnup

2.1. Main characteristics of the primary circuit

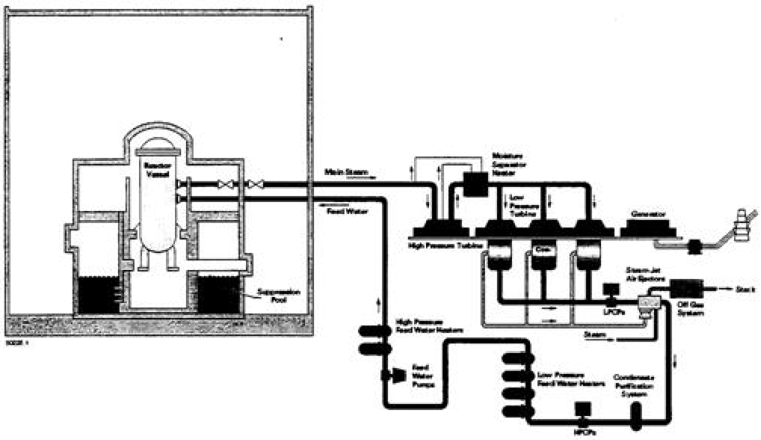

FIG. 5. ABWR – Steam cycle

RMWR is a BWR-type reactor with the innovative MOX fueled core introduced in the ABWR system framework. The primary circuit and its main characteristics are the same as in ABWR shown Figure 5, because the plant system of RMWR is proposed to be the same as that of ABWR except for the reactor pressure vessel (RPV) part. The primary functions of the nuclear steam cycle system are:

1. to deliver steam from the 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 the loss of coolant accident (LOCA) where the RPV does not depressurizes rapidly.

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 the feedwater from the condensate and feedwater system to the RPV.

There are two main steam isolation valves (MSIVs) welded into each of the four MSLs, one inner MSIV inside the containment and one outer MSIV outside the containment. The nuclear pressure relief system consists of safety/relief valves (SRVs) located on the main steam lines (MSLs) between the RPV and the inboard MSIV. The SRVs are designed to provide three main protection functions: overpressure safety, overpressure relief, and depressurization operation, which is discussed below separately.

The automatic depressurization subsystem (ADS) consists of a part of the 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 over-pressure relief logic signals.

2.2. Reactor core and fuel design

The RMWR core configuration consists of 900 hexagonal bundles as shown in Figures 6 and 7. The rated core power is 3,926 MWt (1,356 MWe), which corresponds to 73 kW/l power density. A Y-shaped control rod is adapted in the RMWR core instead of the cross-shaped one used in BWRs, because of better geometrical matching with the hexagonal fuel assembly design. The fuel assembly is in the triangular tight-lattice configuration and contains 217 rods. In order to attain the high conversion ratio, it is necessary to significantly reduce the water to fuel volume ratio in the core. The effective ratio considering the void fraction is less than 0.2 in the present design and about one tenth of that for the ABWR.

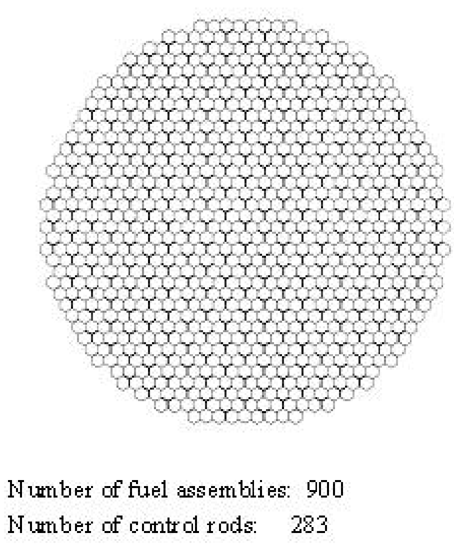

FIG. 6. Core configuration

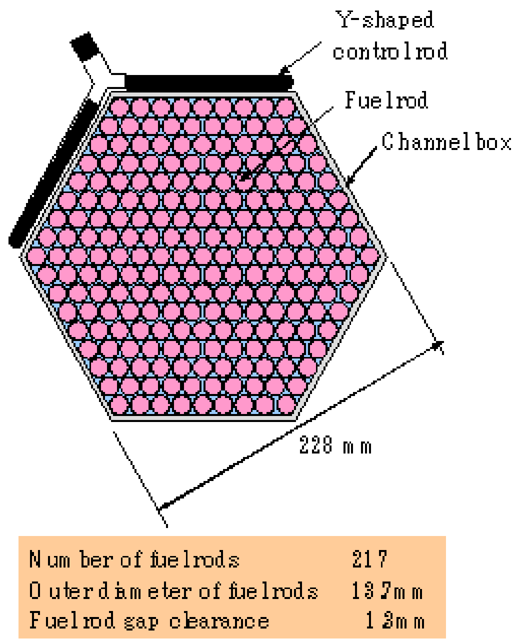

FIG. 7. Fuel assembly

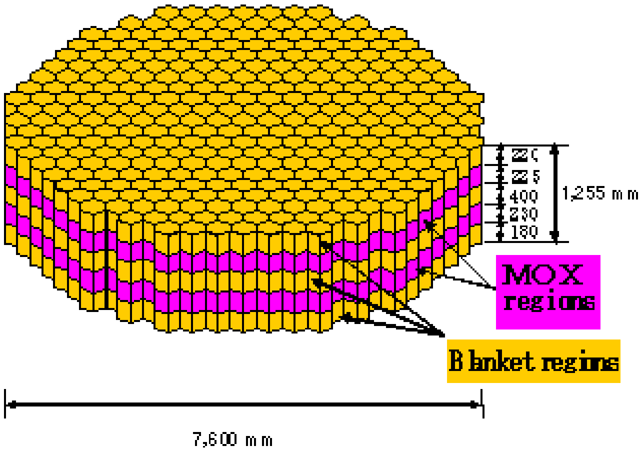

FIG. 8. Schematic of axial core configuration

Although the ABWR utilizes reactor internal pumps (RIPs) to control the recirculation flow through the core, the present design for the RMWR adopted a natural circulation core cooling system, eliminating RIPs and expanding core region. The reactivity control is maintained only by the control rod position.

The fuel rod is 13.7 mm in diameter and is arranged in the triangular tight-lattice configuration with the gap width of 1.3 mm between rods. There is plutonium content distribution in five groups from 12.5 to 19.2 wt% across the assembly to flatten the local power distribution in the assembly. The average plutonium content in the assembly is 18 wt% and the local power peaking factor is less than 1.05. The axial distribution of MOX in a fuel rod is not homogeneous. There are two MOX regions and three blanket regions of the depleted uranium as shown in Fig. 8. There is an inner blanket between two MOX regions, and there are the upper and lower blanket regions.

2.2.1 Control rod drive system

The control rod drive (CRD) system is considered to be the same as for the ABWR, *i.e.* the fine motion control rod drive (FMCRD). It is composed of three major elements: the FMCRD mechanisms; the hydraulic control unit (HCU) assemblies, and the control rod drive hydraulic (CRDH) subsystem. The FMCRDs 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).

However, there are some differences in the control rods from the ABWR design. The control rod is the Y-shaped one as shown in Figure 7 instead of the cross-shaped one. The control rod material is the enriched boron with the high enrichment of 90 %. And, there exists follower above the control rod made of graphite material to reduce the water region.

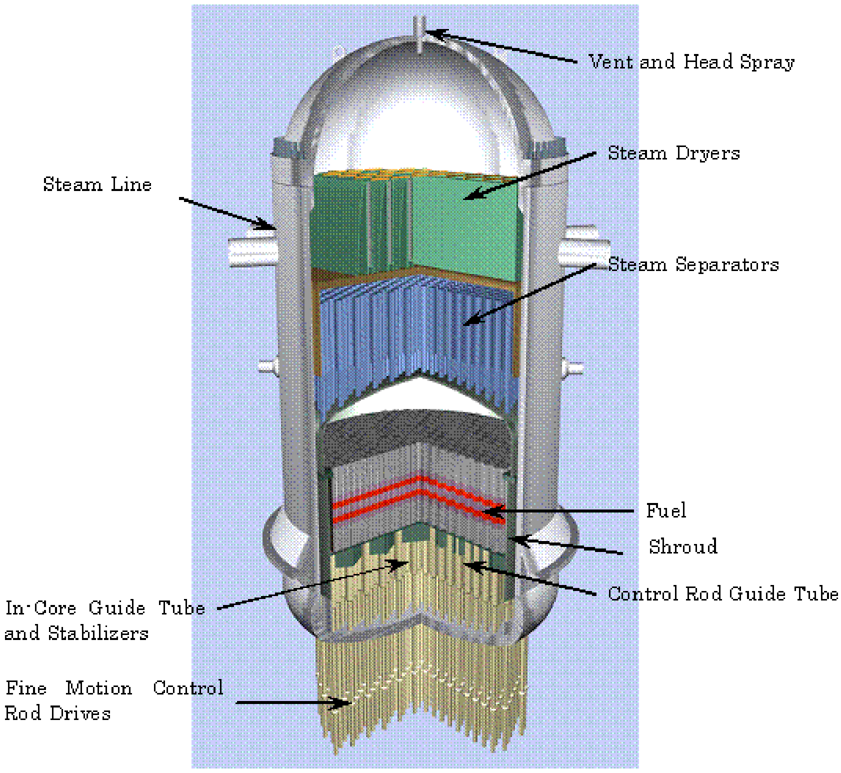

FIG. 9. Reactor pressure vessel and internals of RMWR

2.3. Fuel handling systems

Fuel handling and transfer system are also intended to be the same as in the ABWR system.

2.4. Primary circuit component description

2.4.1 Reactor pressure vessel

The reactor pressure vessel (RPV) system is basically the same as in the ABWR except for the large diameter for the flat core of RMWR.

The RPV 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) 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.

2.4.2 Reactor internals

The RMWR RPV and internals are basically the same as in the ABWR, which are illustrated in Figure 9 except for RIPs. 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.

2.4.3 Reactor recirculation pumps

In the representative design with 1,356 MWe power output, the natural circulation core cooling is adopted.

2.5. Auxiliary systems

The auxiliary is basically the same as in the ABWR. The main auxiliary systems in the 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.

2.6. Operating modes

The RMWR design incorporates the extensive automation of the operator actions which are required during a normal plant startup, shutdown and power range maneuvers. It is basically similar to the ABWR design. The automation features adopted are designed for enhanced operability and improved capacity factor, relative to conventional BWR designs. However, the extent of automation implemented has been carefully selected as in the ABWR 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 control room design is also the same as for the ABWR.

The 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.

2.7. Standard Fuel cycle

The conversion ratio of RMWR exceeds 1.0. The plutonium quality (isotopic composition) of the spent fuels is not deteriorated compared with that of the fresh fuels. Hence, multiple recycling of plutonium is possible and a closed cycle is assumed.

2.8. Alternative Fuel options

RMWR uses light water as a coolant. If for some reason MOX fuel fabrication or reprocessing is stopped and MOX fuel supply is cut off, the core can be operated by replacing MOX fuel assemblies with enriched UO_2 fuel assemblies.

2.9. Spent nuclear fuel and disposal plans if any

The spent MOX fuels are reprocessed and the extracted plutonium is reused in the reactor as the new MOX fuels for itself.

3.1. Safety concept and design philosophy

Safety concept of RMWR follows that of ABWR. The coolant void reactivity coefficient is small and negative. Therefore, the system is robust against events that involve introductions of cold water into the core. Also, there are no RIPs and the core is cooled by natural circulations, so there are no flow reduction events.

The development of the RMWR reactor systems ranges from those for the near-term deployment to those for the long-term. The RMWR for the long-term deployment systems utilize advanced technologies such as fully passive safety systems. In this report, only the RMWR for the near-term deployment system will be described.

The development of RMWR and HC-FLWR aims at the deployment as reactors to replace the current LWRs. Considering this short schedule, it is preferable that the design does not require extensive R&D efforts and significant changes of the current safety regulations in Japan. The safety design for RMWR is, therefore, based on well-matured technologies accumulated for the current generation LWRs especially for ABWR.

3.2. Provision for simplicity and robustness of the design

RIP is eliminated from the system and it is free from core flow reduction events caused by failures of active components. The core cooling and coolant circulations are provided by natural circulations at all times.

3.3. Safety systems and features (active, passive and inherent)

The reactor internal pumps installed in ABWR are eliminated in the RMWR system. Therefore, the passive natural circulation core cooling system is adopted. Another difference between ABWR and RMWR is the use of a passive containment cooling system (PCCS) for the accident management measures, for which a large-scale confirmatory tests were conducted by JAERI.

The ECCS is a three-division system, with a high and low pressure injection pump and heat removal capability in each division functioning independently. One of the systems serves as the reactor core isolation cooling (RCIC) system, which has a steam driven high pressure pump. The ECCS adopts three on-site emergency diesel-generators to support core cooling and heat removal if off-site power is lost. The ECCS is designed to maintain core coverage for any postulated line break size during accidents.

The ABWR has the FMCRD, which adopt a diversified control rod drive mechanism with a fine-tunable electric drive system in addition to the current hydraulic drive system. Response to anticipated transients without scram (ATWS) was improved by the FMCRD, which allow reactor shutdown either by hydraulic or electric insertion.

A Reinforced Concrete Containment Vessel (RCCV) used for the reactor containment for the ABWR is also adopted in RMWR. Employment of the natural circulation core cooling eliminates piping for the reactor coolant recirculating system. This enables the center of gravity of the reactor building to be lowered. Moreover, the flat core of the RMWR makes the building more resistant to earthquakes than the current ABWR.

3.4. Defence in-depth description

The defense in-depth of RMWR follows that of ABWR.

3.5. Safety goals (core damage frequency)

The system behavior prior to core melt is similar to that of ABWR. Since the estimated core damage frequency (CDF) of ABWR is in the order of 10^{-7}/ reactor-years, RMWR can easily attain CDF below 10^{-6} / reactor-years.

3.6. Safety systems to cope with Design basis accidents

Comparable to ABWR.

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

For the mitigation of the effects of the severe accidents, several accident management (AM) measures are planned as done for ABWR. In addition to those AM measures, PCCS is utilized to prevent the containment damage caused by the over-pressurization due to the steam generation during LOCAs. The PCCS is a passive cooling system without relying on the pump operation. It is designed to have sufficient cooling capability for steam condensation with the conservatively estimated amount of noncondensables (nitrogen and/or hydrogen).

The PCCS heat exchanger (HEX) is submerged in the water pool located outside the containment, and is connected to the drywell for the inlet side and the suppression chamber for the outlet side. This PCCS is characterized by the use of horizontal tubes for the HEX, compared to those for the SBWR with vertical tubes. The use of the horizontal HEX has several advantages over the vertical one, which includes the enhancement of the earthquake resistance, the reduction of the pool water level, the easiness of the maintenance. Among the other, the horizontal HEX can be economically optimized, while the optimization is impossible for the vertical HEX because the tube length is limited by the pool liquid level. Large-scale tests were conducted at JAERI to confirm the effectiveness of the horizontal PCCS, from which the promising results have been obtained.

3.8. Provisions for safety under seismic conditions

Seismic isolation technologies are planned to be introduced in the Japanese next generation LWR development project. In this project, various tests are planned for large-scale earthquake isolation equipments that can be applied to nuclear power stations. When these technologies are put to practical use, RMWR can also adopt them.

3.9. Probabilistic risk assessment

Comparable to ABWR.

4.1. Technical features to facilitate implementation of safeguards

Comparable to ABWR.

4.2. Intrinsic features

Instead of blanket fuel assemblies which consist of solely blanket fuels, RMWR core is loaded with fuel assemblies that have axial blanket regions. Therefore, extracting weapon-grade plutonium from the spent fuel is difficult when assembly-wise reprocessing is done. Hence, nuclear proliferation resistance of RMWR is higher compared with FBR which utilizes radial blanket assemblies. Furthermore, multiple recycling of plutonium with low decontamination fuels is possible by small alternations of the axial core compositions. In this case, the isotopic composition of fissile plutonium is reduced and the spent fuel is mixed with minor actinides (MAs) and fission products (FPs). Therefore, nuclear proliferation resistance of RMWR can be further increased.

5.1. Features against human-induced malevolent external impacts and insider actions.

Comparable to ABWR.

6.1. Turbine generator description

The turbine generator plant design is also the same as for the ABWR.

6.1.1 The main turbine

The main turbine is a six flow, tandem compound, single reheat machine. The turbine has one duel-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 at 100 percent nuclear boiling rate.

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 cool-down operations.

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 multipressure three-shell type deaerating type 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.

6.2. Feed water systems

The condensate and feedwater system 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 pumps section. The reactor feed pumps 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 feedwater lines transport feedwater from the feedwater pipes in the steam tunnel through CV penetrations to horizontal headers. Isolation check valves are installed upstream and downstream of the CV penetrations and manual maintenance gate valve are installed 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 transfers this heat to the turbine building service water (TBSW) system. The TBSW system transfers the heat taken from the TBCW system to the power cycle heat sink which is part of the circulating water system.

Electrical and I&C systems

7.1. Design concept

The RMWR system design is based on the well-developed current LWR technologies. The system developed for ABWR can be applicable to that of the RMWR. The instrumentation and control system, including the reactor protection system, can also be applicable, so that the similar configuration to that of the ABWR is to be adopted.

7.2. Power supply systems

On-site power is supplied from either 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.

7.2.1 Direct current power supply

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

7.2.2 Instrument and control power supply

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

7.3. Safety related electrical systems

7.3.1 Class 1E 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 which 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 lE 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.

7.3.2 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. This sizing of the division I battery also meets the requirement to permit operation of the station blackout coping systems. The division I battery is sized to support operation of RCIC and remote shutdown system (RSD), as well as a minimum necessary emergency lighting. This manual load shedding takes credit for the RCIC operation from outside the main control room.

7.3.3 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 uninterruptable, regulated AC power to those loads which require continuity of power during a loss of preferred power (LOPP).

7.4. Control room layout

The control room layout of RMWR is almost the same as for the ABWR.

7.5. Reactor protection and other safety systems

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 insertion of control rods to scram the reactor when unsafe conditions are detected. The RPS uses the function of SSLC to perform its functions.

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 feedwater temperature is different from that for ABWR due to the difference of recirculation flow ratio.

One of the essential components required for instrumentation is the neutron monitoring system. This system is to be composed from the startup range neutron monitoring (SRNM) subsystem, the power range neutron monitoring (PRNM) subsystem, the automatic traversing in-core probe (ATIP) subsystem and the multi-channel rod block monitoring (MRBM) subsystem same as the ABWR.

Spent fuel and waste management

8.1. Provisions for low consumption of non-renewable resources

In RMWR, depleted uranium is used for the base material of the MOX fuels and the blanket regions, and enriched uranium is not used. The representative design of RMWR achieves high burnup for effective use of resources. The average discharge burnup of the inner blanket and MOX regions is 65 GWd/t, while it is 50 GWd/t even when the upper and lower blanket regions are included. Furthermore, the conversion ratio exceeds 1.0 and the plutonium quality does not deteriorate, enabling multi recycling of plutonium by itself. Hence, a sustainable nuclear cycle can be established without consuming natural uranium resources.

8.2. Provisions for acceptable or reduced dose limits

Radioactivity and heat generations of RMWR spent fuels are lower than those of full-MOX LWR with average discharge burnup of 45 GWd/t and Na-cooled FBR with average discharge burnup of 115 GWd/t. They are comparable to those of PWR with average discharge burnup of 45 GWd/t and sufficiently lower than the acceptable limit.

8.3. Provisions for low spent nuclear fuel (SNF) and waste management costs

Spent fuel is recycled. Waste is reduced by reprocessing.

Plant layout

The plant layout is also the same as for the ABWR.

9.1. Buildings and structures

The RMWR plant includes all buildings which are dedicated exclusively or primarily to housing systems and the equipment related to the nuclear system or controls access to this equipment and systems. There are five such buildings within the scope:

- Reactor building - includes the reactor pressure vessel, containment, and major portions of the nuclear steam supply system, refueling area, diesel generators, essential power, non-essential power, emergency core cooling systems, IWAC and supporting systems;
- Service building - personnel facilities, security offices, and health physics station;
- Control building - includes the control room, the computer facility, reactor building component cooling water system and the control room HVAC system;
- Turbine building - houses all equipment associated with the main turbine generator. Other auxiliary equipment is also located in this building;
- Radwaste building - houses all equipment associated with the collection and processing of solid and liquid radioactive waste generated by the plant.

The site plan of the RMWR includes the reactor, service, control, turbine, radwaste and supporting buildings.
Development of the RMWR plant and building arrangements has been guided by the following criteria:

- Retain the passive and well established pressure suppression containment technology;
- Emphasize optimal layout of systems to improve personnel access and equipment maintenance activities.

9.2. Reactor building

The RMWR integrated reactor building and containment structure has been analyzed for a safe shutdown earthquake of 0.3g.

Key distinguishing features of the RMWR reactor building design include:

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

9.3. Containment

The RMWR pressure suppression primary containment system is designed to have the following functional capabilities:

- The containment structure is designed to maintain its functional integrity during and following the peak transient pressures and temperatures which would occur following any postulated loss of coolant accident (LOCA). A design basis accident (DBA) is defined as the worst LOCA pipe break (which leads to maximum containment pressure and/or temperature), and is further postulated to occur simultaneously with a loss of off site power (LOOP) and a safe shutdown earthquake (SSE).The containment structure is designed for the full range of loading conditions consistent with normal plant operating and accident conditions including the LOCA related design loads.
- The containment structure and isolation, with concurrent operation of other accident mitigation systems, is designed to limit fission product leakage during and following the postulated design basis accident (DBA) to values less than leakage rates which would result in off-site radiation doses greater than those set forth in 10 CFR 100.
- Capability for rapid closure or isolation of all pipes or ducts which penetrate the containment boundary is provided to maintain leakage within acceptable limits.
- The containment structure is designed to withstand coincident fluid jet forces associated with the flow from the postulated rupture of any pipe within the containment.
- The containment structure is designed to accommodate flooding to a sufficient depth above the active fuel to permit safe removal of the fuel assemblies from the reactor core after the postulated DBA.
- The containment structure is protected from or designed to withstand hypothetical missiles from internal sources and uncontrolled motion of broken pipes which could endanger the integrity of the containment.
- The containment structure is designed to provide means to channel the flow from postulated pipe ruptures.
- The containment system is designed to allow for periodic tests at the calculated peak or reduced test pressure to measure the leakage from individual penetrations and isolation valves, and the integrated leakage rate from the structure to confirm the leak-tight integrity of the containment.
- The atmospheric control system (ACS) establishes and maintains the containment atmosphere to less than 3.5% (by volume) oxygen during normal operating conditions. To assure an inert atmosphere, operation of two permanently installed recombiners can be initiated on high levels as determined by the containment atmospheric monitoring system (CAMS).

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 RMWR turbine building makes a significant contribution to capital cost savings and a shorter construction schedule.

Plant performance

10.1. Plant Operation

Comparable to ABWR.

10.2. Reliability

The plant system of RMWR is the same as that of ABWR. Hence, the reliability of RMWR plant system is expected to be comparable to that of ABWR.

10.3. Availability Targets

The representative RMWR core design has an operation period of 15 months. The capacity factor of about 92% is expected by assuming 41 days for maintenances and inspections.

10.4. Provision for reduced capital and construction costs

Since the RMWR is a BWR-type reactor with the innovative MOX fueled core introduced in the ABWR system framework, the main design measures aimed to improve the plant economics are the same as the existing ABWR or the future designs (Ref. 1-3), because the plant system of the RMWR is proposed to be the same as that of the ABWR except for the reactor pressure vessel (RPV) part. Those features are advanced-type control-rod drive mechanism (FMCRD), integrated digital instrumentation and control system, large capacity and high efficiency turbine system and so on. Operation experiences accumulated in Kashiwazaki-Kariwa Nuclear Power Station units No.6 & 7 are also beneficial.

In the reference design of RMWR, the difference from the ABWR plant is that the natural circulation core cooling system is adopted and the reactor internal pumps are not installed. That can be possible due to relatively small pressure loss through the core presently designed. The elimination of RIP simplifies the design and reduces the cost.

Thus, construction cost of RMWR is evaluated by the designer to be 97% of that of ABWR. On the other hand, currently the next generation LWR development project is being carried out in Japan. In this project, the target construction period is reduced to 30 months by adopting steel plate reinforced concrete structures, modular fabrications at factories. Introduction of these techniques reduces the on-site works to simple assembling and welding works. These construction techniques can also be adopted for RMWR. In this case, it is expected by the designer that its construction cost can be greatly reduced and the works necessary for maintenance are roughly expected to be halved.

10.5. Construction schedule

Comparable to ABWR. The advanced construction techniques of next generation LWR development project for reducing construction period can be adopted.

10.6. Provision for low fuel reload costs

Since the RMWR aims at multiple recycling of plutonium for the long-term energy supply with the uranium resources, reduction of the fuel cycle cost is important. There are two measures for it. One is to increase the burn-up as much as possible. The other is to reduce the reprocessing cost. A simplified PUREX type reprocessing process has been proposed by JAEA, eliminating the purification processes for uranium and plutonium after their separation process, but keeping the decontamination factor of about 10^5 and reduce the reprocessing cost approximately one half.

Development status of technologies relevant to the NPP

11.1. List of technologies to be included

Tight-lattice fuel assembly cooling technologies (Ref. 4-8)

Deployment status and planned schedule

Concepts of RMWR have been developed by JAERI since 1997 in collaboration with JAPC and Japanese vendors. Concepts of HC-RMWR, which have more conventional core designs than RMWR, have been developed by JAEA since 2004 to be introduced prior to RMWR.

The R&D activities including core and system design studies, thermal-hydraulic experiments, reactor physics experiments, and safety analyses are under way.

Firstly, early introduction and establishment of HC-FLWR as a leading reactor of RMWR is aimed. HC-FLWR has similar design features to those of RMWR, such as triangular-tight fuel rod lattice, hexagonal fuel assembly, Y-shaped control rods. However, the design conditions are brought closer to those of current LWRs by reducing the conversion ratio to about 0.85. Hence, HC-FLWR is a core design concept which can be introduced as replacements for current LWRs. In the early introduction stage of HC-FLWR, the basic technologies are to be established. Namely, theses are the establishments of high-void operation with natural circulation cooling in tight-lattice bundle core and comprehensive integrities of a tight-lattice fuel assembly. After the establishments, the fuel assemblies in HC-FLWR core are successively replaced by the fuel assemblies for RMWR. When all fuel assemblies are replaced, the core operates as an RMWR core.

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Technical data

General plant data

Reactor thermal output	3926 MWth
Power plant output, gross	1356 MWe
Power plant output, net	1300 MWe
Power plant efficiency, net	33.1 %
Mode of operation	Baseload
Primary coolant material	Light Water
Moderator material	Light water
Thermodynamic cycle	Rankine
Type of cycle	Direct
Non-electric applications	Desalination

Safety goals

Core damage frequency <	10E-6 /Reactor-Year

Nuclear steam supply system

Steam flow rate at nominal conditions	2122 Kg/s
Steam pressure	7.2 MPa(a)
Feedwater flow rate at nominal conditions	2118 Kg/s
Feedwater temperature	277 °C

Reactor coolant system

Primary coolant flow rate	6500 Kg/s
Reactor operating pressure	7.2 MPa(a)
Core coolant inlet temperature	283 °C
Core coolant outlet temperature	288 °C
Mean temperature rise across core	5 °C

Reactor core

Active core height	1.255 m
Equivalent core diameter	7.2 m
Average linear heat rate	16.0 KW/m
Average core power density	73.3 MW/m^3
Fuel material	UO2 and MOX
Cladding material	Zircaloy-4
Outer diameter of fuel rods	13.8 mm
Lattice geometry	Triangular
Number of fuel assemblies	900
Enrichment of reload fuel at equilibrium core	11.4 Weight %
Fuel cycle length	15 Months
Average discharge burnup of fuel	50 MWd/Kg
Control rod absorber material	B4C

Reactor pressure vessel

Inner diameter of cylindrical shell	8900 mm
Wall thickness of cylindrical shell	170 mm
Design pressure	8.62 MPa(a)
Design temperature	302 °C
Base material	Stainless Steel
Total height, inside	19400 mm

Primary containment

Overall form (spherical/cylindrical)	Cylindrical
Dimensions - diameter	29 m
Dimensions - height	36 m
Design pressure	310 MPa

Design temperature	171 °C
Design leakage rate	0.4 Volume % /day

Safety injection systems

Active/passive systems	Active

Turbine

Type of turbines	Six flow, Tandem compound, Single reheat
Turbine speed	1500 rpm
HP turbine inlet pressure	6.79 MPa(a)
HP turbine inlet temperature	284 °C

Generator

Rated power	1540 MVA
Active power	1356 MW
Voltage	27 kV
Frequency	50 Hz

Condenser

Type	Shell type
Condenser pressure	11.75 kPa

Feedwater pumps

Number	3
Pump speed	5000 rpm
Head at rated conditions	600 m
Flow at rated conditions	1 m^3/s