Status Report – APWR (Mitsubishi, Japan)

Overview

Full Name:	**Advanced Pressurized Water Reactor**
Acronym:	**APWR**
Reactor type:	**Pressurized Water Reactor**
Coolant:	**Light water**
Moderator:	**Light water**
Neutron spectrum:	**Thermal neutrons**
Thermal capacity:	**4466 MWth**
Gross electrical capacity:	**1538 MWe**
Design Status:	
Designer:	**Mitsubishi**
Last update:	**May 2012**

Description

APWR (Mitsubishi, Japan)

1. Introduction

Nuclear power generated by light water reactors accounts for approximately 1/3 of Japan's power supply. Also, it is expected to play an important role in providing energy security and preservation of the global environment in the future.

- The advanced PWR (APWR) has been developed, as a nuclear power plant for future use in Japan, as a joint international cooperative development project by seven companies comprising the five PWR electric power companies (Hokkaido, Kansai, Shikoku, Kyushu Electric Power Company, and Japan Atomic Power Company) and Mitsubishi Heavy Industries and Westinghouse. Its development was part of Phase III of the Improvement and Standardization Program of Japan's Ministry of International Trade and Industry (Currently the Ministry of Economy, Trade, and Industry).
- In the APWR, advanced technologies based on the operational experience gained so far have been incorporated. Also safety, reliability, operability and the performance of the plant have been further increased, and the construction cost has been further reduced due to the benefit of economy of scale resulting from the increase in capacity.

- The first APWR plant will be adopted by the Japan Atomic Power Company, Tsuruga-3 and 4. In this document some outstanding features of this new APWR are described. Summary level Technical Data are provided in the Appendix.

2. *Description of the Nuclear Systems*

- Primary circuit and its main characteristics
  - Table 2-1 shows a comparison of some major parameters between the APWR and an existing four-loop plant in Japan. The APWR is in the largest capacity class of LWRs in Japan and has adopted, for example, high performance steam generators and low pressure turbines with approximately 54 in. (1375 mm) last stage blades. Various improvements have been incorporated in the reactor core so that operation with long fuel cycles is possible using low enriched fuel in order to reduce uranium requirements, and to provide increased flexibility for various application such as the use of MOX cores and high burn-up fuels.
  - Also critical equipment such as reactor internals and steam generators have been designed taking into account operational experience of aging on operating plants so that a high degree of reliability can be obtained. To ensure safety, the reliability of the equipment and systems has been increased, and highly advanced safety systems such as methods of providing assistance to the operations during abnormal events have been adopted.
  - For instrumentation and control systems, the latest digital control technologies have been incorporated not only in the reactor control system but also in the reactor protection system, and also the latest electronics technologies to improve the man machine interface have been introduced in the main control room.
  - In addition, in order to make the plant easier to maintain, a variety of improved technologies have been inconporated, thus improving the efficiency of periodical inspections and reducing exposure to employees.

- Reactor core and fuel design
  - The reactor core, consisting of 257 17×17 fuel assemblies, has a core thermal output of approximately 4,451 MWt. Considering the need to reduce fuel cycle costs and future needs for MOX reactor cores and high burn-up cores, a variety of improvements have been incorporated in the reactor core. Also the core has been designed so that it can use plutonium fuel with 1/3 or more MOX cores, and is flexible enough to use fuel with a burn-up of long operating cycles.
  - To reduce fuel cycle costs, the fuel assemblies have zircalloy grids with low neutron absorption and the core is surrounded with a reflector to reduce neutron leakage, thus increasing neutron efficiency.
  - The reactor uses the 17×17 fuel which has operated well in existing Japanese plants. The design is made by adopting zircalloy grids with low neutron absorption as above-mentioned and other means, in order that it can be used for high burn-ups and increased loadings of MOX fuel.

- Fuel handling systems
  - The fuel handling and fuel transfer systems cousist mainly of the refuelling crane, fuel transfer system, and multi-functional mast type spent fuel pit crane as in existing Japanese plants.

- Considering the recent need to reduce the periodical inspection time, many improvements including an increase in the speed of each system have been introduced.
- Also, in order to reduce operations in radiation controlled areas, these systems can be operated automatically from a remote centralized control room instead of the present method of operating individually from a control station next to each piece of equipment.

TABLE 2-1. MAJOR APWR PARAMETERS

	APWR	Existing four loop PWR in Japan
Electric power output	1,538 MWe	1,180 MWe
Core thermal output	4,451 MWt	3,411 MWt
Reactor thermal output	4,466 MWt	3,423 MWt
Fuel type	17×17	17×17
Number of fuel assemblies	257	193
Fuel effective length	Approximately 3.7 m	Approximately 3.7 m
Total uranium inventory	Approximately 121 tonnes U	Approximately 89 tonnes U
Number of control rods	69	53
Neutron reflector	Stainless steel	None
Reactor vessel	Approximately 5.2 m inner dia. & Approximately 14 m height	Approximately 4.4 m inner dia. & Approximately 13 m height
Steam generators	70F-1 type	52F type
Primary coolant pumps	MA25S type	93A-1 type
Primary system flow (m^3/h/loop)	Approximately 25,800	Approximately 20,100
Turbine	TC6F54	TC6F44
Generator	1,715 MVA	1,310 MVA
Containment	PCCV	PCCV
Engineered safety systems	Four trains of mechanical systems & Two trains of power systems	Two trains
Refuelling water storage pit	Inside containment	Outside containment
Reactor protection system	Digital	Analog
Reactor control system	Digital	Digital
Main control room	Improved	Standard

- Primary circuit components description
  - Reactor pressure vessel
    - Although the reactor vessel inside diameter has been increased to 5.2 m in order to accomodate 257 fuel assemblies, the vessel is made with forged rings and has no longitudinal welds in the core area as is the case with the latest four-loop plant. The neutron irradiation of the steel opposite the core has been reduced to approximately 1/3 as compared with the 4-loop plant reactor vessel with a 40-year neutron fluence of about 2×10^{19} n/cm^2 by providing a neutron reflector, thus increasing the reliability of the reactor vessel.
    - Also, in order to reduce the susceptibility to corrosion cracking of the penetrations of the reactor vessel head, the nozzle material has been improved (from alloy 600 to heat treated alloy 690) and the primary coolant temperature in the top dome of the reactor vessel is designed to be reduced to reactor inlet temperature.
    - A comparison between the APWR and an existing four-loop plant is shown in Figure 2-1.

  - Reactor internals
    - The reactor internals are increased in the radial dimensions of the members according to an increase in size of the core. The neutron reflector consisting of eight rings of stainless steel blocks not only reduces fuel cycle costs but also reduces the irradiation of the reactor vessel and core internals. By installing the neutron reflector, the neutron irradiation of the reactor vessel can be reduced to approximately 1/3 compared to present Japanese reactors. On present reactors, the core baffle is a plate structure held together with 2000 or more bolts, whereas the new neutron reflector has a simple construction which does not use bolts in the effective core area.
    - The improved core internals underwent flow tests to check the validity of the design.

  - *Steam generators*
    - The APWR has adopted steam generators (Type 70F-1) to match the increased capacity of the reactor core. The tubes are 3/4 in (19 mm) diameter which is smaller than the 7/8 in. (22 mm) used in existing Japanese plants. This results in a more compact steam generator with resistance to earthquakes, etc.
    - The tubes of the steam generators are made of thermally treated alloy 690 (TT690). Also the design of the anti-vibration bars in the U-bend area of tubes has been improved to reduce the risk of flow induced vibration of the tubes.
    - In addition, in order to make maintenance and inspections easier, accessibility has been improved by increasing the diameter of the manholes and in other ways. Figure 2-2 shows a schematic drawing of the steam generator.

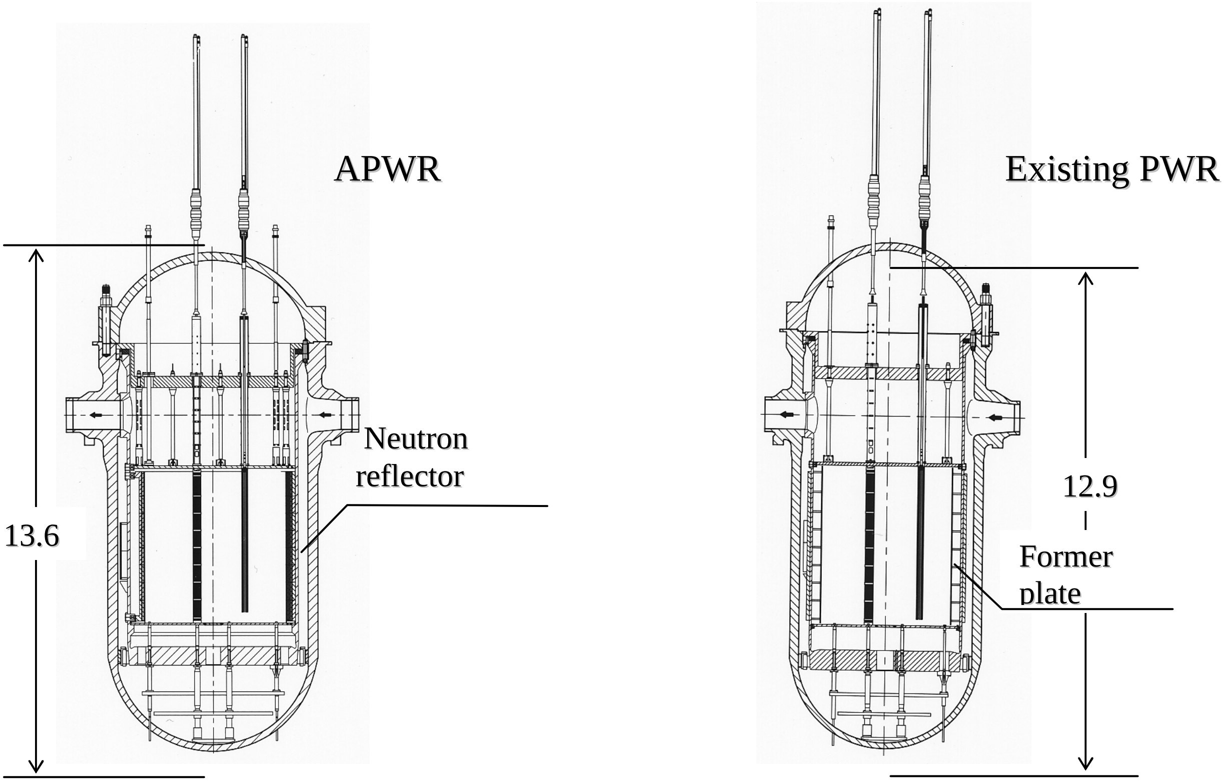

FIG. 2-1. A comparison between the APWR and an existing Japanese PWR

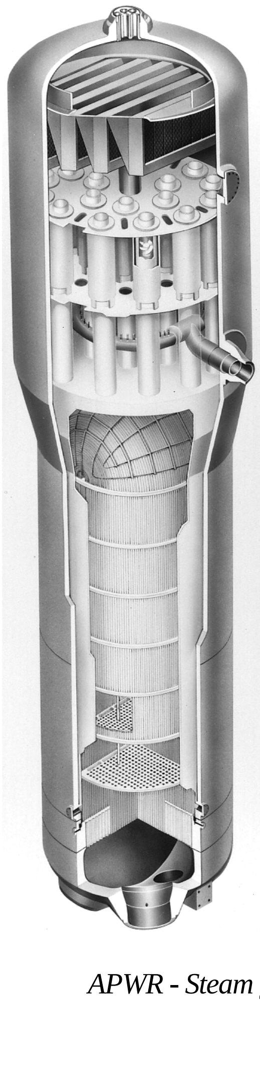

FIG. 2-2. APWR - Steam generator

- *Reactor coolant pumps*
  - Because the primary coolant pump has to supply a flow approximately 30% larger than that of an existing primary coolant pump, a MA25S type pump (60 Hz) which is larger in capacity than the existing MA25S type has been adopted. For the improved No.1 seals, heat-resisting 0-rings are employed as well as ceramic material which has larger size and excellent durability, thus aiming at enhancement of the reliability.

- Main coolant lines
  - For piping material, low alloy steel (with stainlesss steel lining) is used from the point of view of enhancement of reliability and inspect ability.

- Reactor auxiliary systems
  - Chemical and volume control system
    - The chemical and volume control system has the following main functions.
    - The first function is to adjust the amount of primary water contained in the reactor coolant system. In normal operation, the letdown and charging flows are controlled so that the water level in the pressurizer is kept at the programmed level. Seal water is injected into the primary coolant pump seals.
    - The second function is to adjust the concentration of boron and the quality of water contained in the primary coolant system. The concentration of boron in the primary coolant system is adjusted by adding pure water from the reactor make up system or boric acid solution as required compensating for fuel burn up.
    - The quality of water in the primary system can be controlled by adding hydrazine or lithium hydroxide, passing the water through a cation demineralizer, and adding hydrogen gas to the vapor space of the volume control tank.
    - The third function is to purify the primary coolant. The primary coolant is purified by a demineralizer and filter in the let down line.
    - The let down flow is taken from the cross-over leg of the primary coolant system, and the coolant is cooled by the regenerative heat exchanger and let down heat exchanger, and then purified in the demineralizer.
    - To supply water to the primary coolant system and seal water to the primary coolant pumps, a charging pump is used taking water from the volume control tank.
    - Boric acid tank water injection via the safety injection nozzle to the core, using two charging/boron injecting pumps as well as insertion of the control rod cluster and operation of the emergency core cooling system (ECCS) make the core subcritical earlier at the time of an overcooling event.

- Operating mode
  - The reactor is designed so that it can be operated automatically within the range of 15 to 100% of rated output by the reactor control system.
  - The reactor control system is designed so that it can accommodate the following load changes without causing a reactor trip.
    - a 10% step load change (within the range of 15 to 100%)
    - a 5% per minute ramp load change (within the range of 15 to 100%) 65% load reduction
  - With respect to the load fluctuation following capacity to the following electric power system, the following is provided:
    - Daily load-follow operation of 100%-50%-100%
    - Automatic frequency control or governor control to control system frequency over a load range of 5%.

3. Description of Safety concept

- Safety requirements and design philosophy
  - The configuration of the ECCS mecanical systems has been changed from the conventional two trains to four trains to give more redundancy and independence. Also, tie lines between trains have been eliminated to simplify the systems and increase the reliability.
  - In existing Japanese plants, refilling of the reactor vessel and reflooding of the reactor core after a LOCA are made by both the accumulators and low presssure injection pumps. In APWR, however, advanced accumulators with two-stage injection characteristics have been adopted and the present low pressure injection systems and the accumulators have been integrated to simplify the equipment and increase the functional reliability. Also, the refuelling water storage pit is installed inside the containment, thus eliminating the operation of changing the suction from the refuelling water tank to the containment recirculation sump which is needed during an accident on existing Japanese plants. A comparison between the ECCS of an existing Japanese plant and the APWR is shown in Figure 3-1.
  - As a result, it has been found by the design organization that, as a whole, the APWR is expected to have a core damage frequency of at least one order of magnitude lower than the existing Japanese 4-loop plant with a core damage frequency of about 10^{-7}/ry.

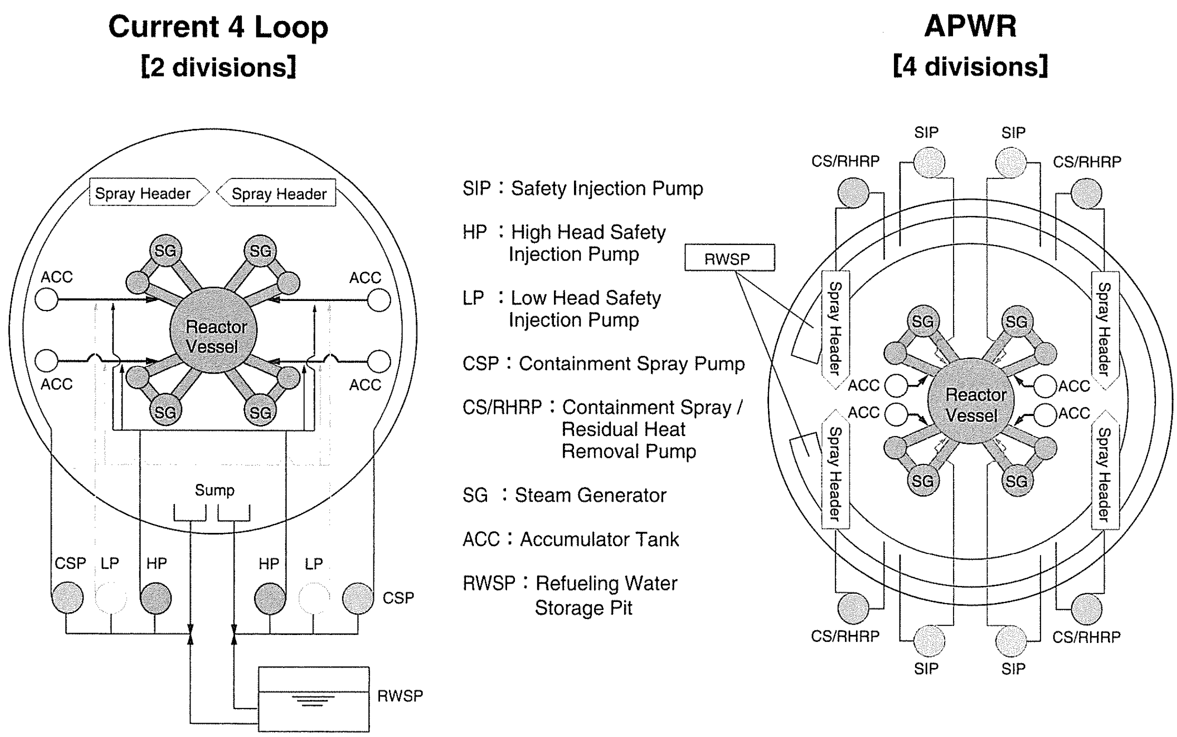

FIG..3-1. APWR emergency core cooling systems compared with existing Japanese PWR

- *Deterministic design basis*
  - The safety design of an APWR satisfies, from a deterministic design point of view, the safety design criteria for design basis events. Also, using probablistic assessments, the APWR is designed so that it has sufficient margins for beyond design basis events. The design basis events are abnormal operating conditions which are classified into two groups: abnormal operating transients and accidents during operation, and safety criteria have been set for each group. The standards for radiation exposure are specified for normal operation and accidents, thus reducing the risk to the general public and employees to less than an allowable limit.

- Risk reduction
  - To further reduce the risk and provide increased protection, the reactor is designed to have a high degree of safety. Specifically, it is designed with the following design targets.
    - The core damage probability during power operation should be used as a quantitative index for the increase in safety. As a target, the probability should be reduced to about 1/10 th of that of the latest Japanese PWR.
    - The core damage frequency during shutdown should be approximately the same as the target for power operation.
    - For further protection, the containment failure frequency (CFF) should be reduced to an appropriate level (to approximately 1/10 of the core damage frequency as a target).

        - The structure of the containment should be designed so that its functions as a target can be maintained for one day or longer during quasi-static pressurization following a severe accident. For pre-mature failure modes caused by missiles and dynamic loads, countermeasures should be taken for containment design, etc.

    - Especially, these measures are concretely classified as follows.
        - Countermeasures against core damage during power operation. Although a sufficiently low core damage frequency can be achieved as a result of the increase in safety provided by the four sub-system safety systems, installation of the emergency water source inside the containment, etc., countermeasures against an interfacing system LOCA and other events have been also taken which further reduce the risk.
        - Countermeasures for increased safety during shutdowns. These include installation of an automatic interlock to isolate the letdown line when the reactor coolant system (RCS) water level is lowered, improvement of water level monitoring, improvement of the RCS water injection function during shutdowns and other countermeasures. These countermeasunes are under consideration.
        - Countermeasures for mitigating the effects of an accident. These include the use of the containment vessel air recirculation systems, alternate sprays supplied from the fire service water systems, countermeasures for hydrogen control, etc., and, at the same time, countermeasures against the events which could become a potential threat to the containment are also studied by water injection into the cavity from the fire service systems, improvement of the cavity shape etc..

- Safety systems and features (active, passive and inherent)
    - Safety systems configuration
      The emergency core cooling systems and containment spray/residual heat removal system consists of four identical and independent mechanical sub-systems. Power is fed from two independent and redundant emergency power systems.
    - The basic configuration is as follows.
        - Four sub-systems each having one safety injection pump, containment spray/residual heat removal pump, and containment spray/residual heat removal cooler;
        - One refuelling water storage pit installed inside the containment;
        - Four advanced accumulator tanks. (Capacity: About 90 m^3/unit)
    - The advanced accumulators refill the reactor vessel lower plenum and downcomer immediately after a LOCA with a medium to large break size and, after that, they inject water to reflood the core and function as both the accumulator tank and low pressure injection pump of existing Japanese plants.
    - The safety injection pump has a function to take water from the refuelling water storage pit installed at the bottom of the containment and feed cooling water to the reactor vessel for a long period after the accident.

- The containment spray/residual heat removal pumps are used as residual heat removal pumps, and also used as containment spray pumps.
- The auxiliary feed water system (AFWS) supplies the auxiliary feedwater required by the steam generator when the normal feedwater system is not available. Except for the auxiliary feed water storage pit, this system consists of two mechanical sub-systems. Each sub-system consists of a motor driven auxiliary feedwater pump and a turbine driven auxiliary feedwater pump.

- Emergency core cooling systems
  - The emergency core cooling system feeds sufficient cooling water into the core in a LOCA situation.
  - When an “S” signal of safety injection is initiated, the safety injection pumps are started automatically to take water from the refuelling water storage pit located in the containment and inject coolant directly into the reactor vessel without passing through the loop.
  - Immediately after the blow-down of the primary coolant during a LOCA caused by a large or medium sized break, the advanced accumulators are used to refill water into the reactor vessel lower plenum and downcomers, and to inject, together with the satety injection pump, cooling water until the core is reflooded. At the start of injection, cooling water is injected with a large flow rate and then, when the water in an accumulator has dropped to a certain level, the flow damper switches over the flow to a smaller flow rate which provides an injection flow similar to that given by the safety injection pumps on current Japanese plants.

- *Containment spray system*
  - Four containment spray/residual heat removal pumps and four containment spray/residual heat removal coolers function as a containment spray system if a LOCA or main steam line break accident occurs.
  - When containment spray “P” signals are initiated, four containment spray/residual heat removal pumps are started automatically, and the stop valves in the pump discharge lines are opened automatically. The containment spray/residual heat removal pumps take water from the refuelling water storage pit and supply it to the containment spray header located at the top of the containment through the containment spray/ residual heat removal coolers.
  - This system has also a residual heat removal function used to remove decay heat from the core in normal cooling of plants and refuelling.

- In-containment refuelling water storage pit
  - The refuelling water storage pit is formed in a horse shoe shape, and is located at the bottom level of the containment.
  - It provides a continuous source of water for the safety injection pumps and containment spray/residual heat removal pumps. Therefore, it is not necessary to switch over from the refuelling water storage pit to the containment recirculation sump as must be done on existing Japanese plants. During refuelling, the refuelling

water storage pit is used also as a water source to fill the reactor cavity.

- Emergency feed water system
  - The auxiliary feed water system, except for the emergency feed water storage pit, consists of two mechanical sub-systems. Each sub-system is provided with a motor driven auxiliary feedwater pump and a turbine driven auxiliary feedwater pump.
  - On receiving signals from the reactor protection system, the system starts feeding water automatically from the emergency feed water storage pit to the steam generator. If the steam generator heat transfer tubes, main feedwater pipes, or main steam pipes are broken, the system isolates the auxiliary feedwater to the damaged steam generator automatically by auxiliary feedwater isolation signals.
  - When the auxiliary feedwater system has been started and the plant has been stabilized at hot standby conditions after an accident or transient, the auxiliary feedwater system can be used to cool the plant to the temperature at which the residual heat removal system can be put in service. When that temperature is reached, the residual heat removal system is started to bring the plant to cold shutdown, and the auxiliary feedwater system is stopped.

- Residual heat removal system
  - The residual heat removal system removes decay heat from the core by taking water from the hot legs of the primary cooling system by the four containment spray/residual heat removal pumps and returning the coolant to the cold legs of the primary cooling system through the four containment spray/residual heat removal coolers. The residual heat removal system has the capacity to cool the primary coolant temperature from 177 to 60°C within 20 hours after the reactor has been shut down.

- Severe accidents (beyond design basis accidents)
  - Prevention of severe accidents
    - In the preliminary design of the APWR, a safety system with four sub-systems has been adopted, the refuelling water storage pit (RWSP) has been located in the containment, advanced accumulators have been included, and the auxiliary feedwater system and auxiliary cooling water system/sea water system have been improved functionally. Thus, a high level of safety has already been provided to ensure core integrity.
    - Regarding the interface LOCA which bypasses the containment, the corresponding parts of the piping in the residual heat removal system are under consideration with a higher rating to prevent the interface LOCA from occurring since this type of accident can have very serious consequences to the environment.

  - Countermeasures during plant shutdowns
    - As safety enhancement during mid-loop operating mode, which are especially important among the countermeasures during plant shutdowns, some countermeasures are to be taken, such as adoption

of RCS high water level operation, reinforcement of RCS water level indicators, automatic isolation of letdown line at low RCS water level, reinforcement of water injection functions during lowering of RCS water level, etc.
  - Also, as a precaution against the event of abnormal dilution of boric acid during the external power failure, interlocks to prevent boron dilution are under consideration.

- Mitigation of severe accidents
  - In an APWR plant, as shown in Section 4.3.6.1, the mitigation of the consequences of a severe accident is also to be considered from the view point of risk reduction and greater protection. Specifically, as shown below, countermeasures against those events that threaten the integrity of the containment vessel are under consideration.
  - As the countermeasures against debris dispersion, the reinforcement of depressurization function of the primary system and the improvement of RV cavity form are considered countermeasures against damage by quasi-static over pressure, the normal containment vessel air recirculation system and an alternative containment vessel spray supplied from the fire service water system can be used. These systems can be used to cool the containment vessel and reduce the pressure if the containment vessel spray system is not available.
  - As countermeasure against containment vessel damage due to hydrogen combustion, a hydrogen control system (ignitors) can be installed to control the hydrogen concentration.
  - To provide adequate cooling of molten debris ejected from the reactor vessel, sufficient floor space will be provided in the RV cavity and water will be injected into the cavity from the fire service water system. Also a 1 m thick protective layer of concrete can be provided so that the containment vessel boundary is not exposed directly to the debris. Thus the molten debris will be coolable, and erosion of the concrete and overheating of the containment vessel atmosphere can be prevented. As countermeasures against the dispersion of debris, reinforcement of the primary system depressurization function and improvement of the reactor vessel cavity form are considered. It is also considered that the outlet from the RV cavity to the other containment vessel spaces should be constructed like a labyrinth.

4. Proliferation resistance

No information is provided now.

5. Safety and Security (physical protection)

No information is provided now.

6. Description of turbine-generator systems

- Turbine generator plant
  - The high pressure turbine and low pressure turbines are double flow turbines with reaction blading.
  - The last stage blades are approximately 54 in. (1375 mm) ISB blades (Integral Shroud Blade) to increase the electric power output and efficiency. The high performance blades, low pressure turbine casing which improves exhoust loss, etc. further increase the efficiency. Also the moisture extraction system has been improved to reduce erosion.
  - The 54 in. last stage blades have been subjected to vibration tests and actual load tests to demonstrate that the turbine has a high performance and reliability.
  - The 1715 MVA generator which is of 4-pole type, has a larger rotor diameter than that of an existing PWR plant in order to increase the output. The rotor windings are cooled internally by hydrogen gas while water is used for cooling of the stator. The excitation is provided by a brushless static system.
  - Figure 6-1 shows a view of the turbine generator unit.

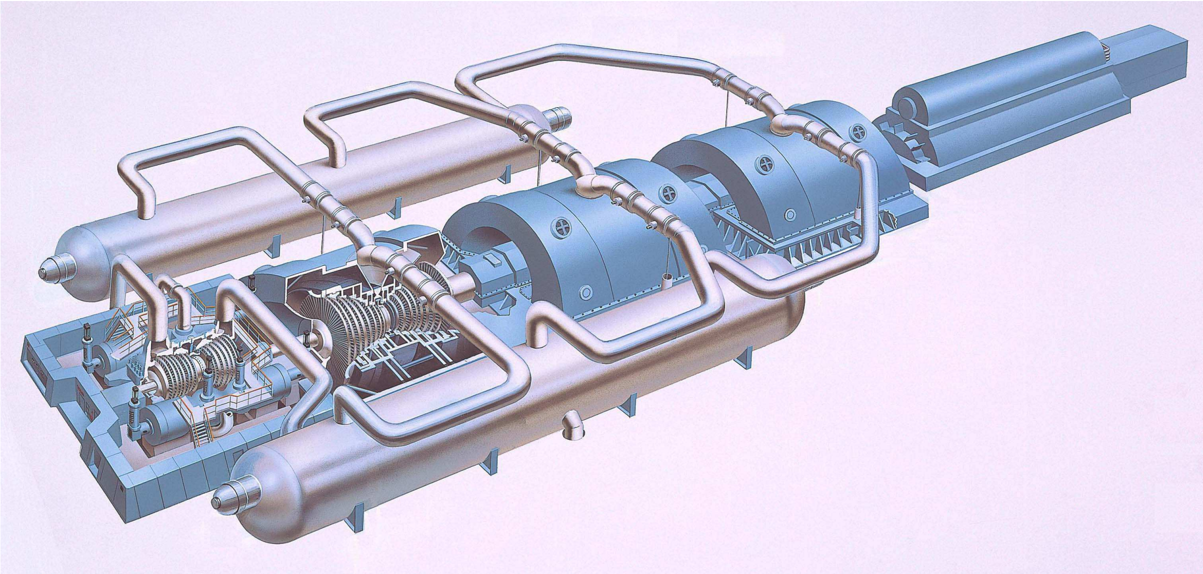

FIG. 6-1. APWR - Turbine generator unit

- Condensate and feed water systems
  - The moisture separator/re-heater has a two stage heater, and can achieve a high efficiency. The turbine building has been reduced in size by reducing the outside dimensions of the moisture separator and adopting the so-called fourneck heater system. In this system four low pressure feed water heaters are installed inside of the condenser, whereas a conventional plant usually has two low pressure feed water heaters in this location.
  - The feed water systems use six extraction stages. These systems consist of 3 trains of four stages of low pressure heaters, the deaerator and 2 trains of single stage high pressure heaters. In particular, the deaerator is installed on the upstream side of the final high pressure feed heater to obtain the benefits of a direct contact heat exchanger.

- The feed water heater tubing material is changed to stainless steel. This is to avoid corrosion of the low pressure feed water heater tubes caused by ammonia and to improve water chemistry.
- The capacity of the pumps installed in the condensate system is 50% $\times$ 2 units, and the capacity of the pumps installed in the feed water system is 50% x 2 units and 40% x 1 unit (backup). Even if one pump fails, partial power operation can be maintained.

- Auxiliary systems
  - The turbine plant has the following additional features.
    - To improve efficiency, the drains from low pressure feed water heaters are collected in the condensate system on the down stream side of the next heater.
    - To further reduce the iron concentration in feed water, low alloy steel in the system equipment, etc. are applied.
  - To simplify the systems and equipment, two circulating water pumps (50% capacity) have been installed in the circulating water system as is the case with the current Japanese PWRs.

7. *Electrical and I&C systems*

- Design concept, including control room
  - The main control room is provided with compact consoles on which CRTs and flat display panels are mounted. Conventional operating and monitoring devices such as switches, lamps, indicators, and recorders have been eliminated.
  - Touch screen operations are applied, and the plant parameters and operating swiches are displayed on the same screens that are used for operating the plant. Therefore, the work load of the operators is reduced and the reliability of operation is increased.
  - On the wall of the main control room, a large display panel is installed to display the major monitoring parameters for normal and abnormal conditions of the entire plant. Thus the current status of the entire plant can be understood by everyone and communication among operators is improved.

- Reactor protection and other safety systems
  - The reactor protection system and other safety systems are digital systems of the functionally distributed type.
  - The reactor protection system consists of four channels including the reactor trip breakers. Each channel is formed with multiple digital devices so as to provide redundant protection functions and to separate the reactor protection function from the other safety system operating functions.
  - The other safety systems consist of two trains. Each train has two sets of digital devices. To interface these systems with the auxiliary equipment in the plant, remote input/output devices arranged distributedly in the plant are connected to the host computer through optical fiber cables, thus reducing the volume of wiring.

- The reactor protection system and other safety systems are provided with automatic test equipment so that periodical tests can be carried out fully automatically.
- To achieve high reliability, the software used for the digital safety systems is modularized and simplified and a symbolic language was used in the design. Verification and Validation tests are to be carried out to the maximum extent possible.

- Operational power supply systems
  - The operational AC power supply system can receive external power from the main power supply system and stand by power supply system. Power from the main power supply system comes through the main transformer and unit transformers. When the plant is operating normally, the main generator is connected to the external power system and, when the main generator is stopped, it is disconnected from the external power system by the generator load break switch.. Therefore, the operational AC power supply system can receive power through the unit transformers continuously whether the main generator is operating or not. If the main power supply system fails and the plant does not continue to operate independently, power will be received through the standby power supply system. Power from the standby power supply system is received through the emergency transformer which has sufficient capacity to enable the plant to be shut down safety.
  - The buses of the operational AC power supply system are divided into two main groups: the 6.6 kV high voltage system and the 440 V low voltage system, each comprising normal buses to supply power to loads such as primary coolant pumps, main feed water pumps and other equipment required for normal plant operation and two-trains of emergency buses to supply power to loads such as safety injection pumps and other equipment required for the safety of the plant.
  - In addition to the above AC power supply systems, other power systems have been provided which can be supplied from batteries in the event of an interruption or station blackout and an instrumentation and control power supply (consisting mainly of inverters) for supplying power to the instrumentation and control equipment which are mainly computer loads.

- Safety-related systems
  - The emergency power systems for supplying power to the operating power systems when an accident occurs in the plant or there is a loss of external power, include the emergency diesel generators and battery equipment. The emergency diesel generators will start automatically immediately after an accident occurs or external power is lost, and supply power to the emergency buses. The emergency power systems are made as redundant systems, and the safety of the plant can be secured with only one train of emergency power.
  - The DC power supply system can supply power to the instrumentation control power system during an instantaneous power failure, and also has sufficient capacity and to supply the switchgear which must operate following a loss of external power and to supply the initial excitation power for the diesel generators. Also it has sufficient capacity for

maintaining the safety of the plant following a total failure of all AC power.

- The bus configuration for the emergency power system is designed so that it is consistent with the configuration of the plant safety systems. As a result, the AC and DC power systems are divided into two trains to be consistent with the two trains and four sub-systems, and the instrumentation and control power system is divided into two trains/four channels to be consistent with the four channels.

8. *Spent Fuel and Waste management*

No information is provided now.

9. *Plant layout*

The plant must be laid out so that the safety of the reactor facilities is not impaired, and the exposure dose around the plant is below a specified limit. Also separation of redundant trains, earthquake resistance, and maintenance of the safety system equipment must be considered to give an optimum arrangement.

- Buildings and structures, including plot plan
  - The standard arrangement is for a twin unit plant consisting of two reactor buildings, a turbine building, a common control building and auxiliary building.

- Design requirements
  - The buildings, structures, equipment, and pipes are classified into the following three classes: S, B, and C. The seismic design must be made according to the class concerned.
    - **Class S**: Those facilities which contain radioactive materials themselves or are associated directly with facilities which contain radioactive materials, and which may release radioactive materials outside the plant if they fail to function properly. Also, facilities which are required to prevent such releases and to reduce the effect of radioactive materials dispersed to the environment if an accident occurs, and those which have serious consequences for the plant.
    - **Class B**: Those which have relatively small effects on the plant compared to Class S.
    - **Class C**: Those for which safety standards equivalent to those of general industrial facilities are adequate and facilities not classified as S or B.
  - Based on the above classifications, the seismic design of the buildings is made by classifying the reactor building and control building into Class S, the auxiliary building into Class B, and the turbine building into Class C.
  - To ensure safety against aircraft impacts, in principle, a site must be selected which is not close to an airport and air route if aircraft impacts are not to be considered in the design.
  - The plant must be designed as follows for internal and external events such as jet aircraft, missiles, and fires.

    - **Internal missiles**: The design must be made in such a way that the safety of the reactor is not impaired due to the effects of internal missiles and broken pipes.
    - **Fire**: To prevent the safety of reactor facilities from being impaired by fire, the plant must be designed using a proper combination of three general rules based on the "Guidance for Verification of Fire Protection of LWR Facilities for Power Generation" in Japan.
      - *(a)* Prevention of fires
      - *(b)* Detection of fires and fire extinguishing
      - *(c)* Reduction of the effects of fires
  - In principle, the structures, systems, and equipment critical for safety must be so designed that the reactor facilities do not make common use of any one of them provided that, judging from the functions and construction, the safety of the reactor may be impaired by such common use.
  - The reactor facilities must be arranged in the plant site of the plant so that the exposure dose to the general public in those areas outside the controlled areas and around the plant is below a specified limit. Furthermore, they must be sufficiently far from the site boundary so that the exposure dose received in areas outside the site during major accidents and hypothetical accidents is adequately below the target dose indicated in the Japanese "Guidance for Verification of Reactor Siting."
  - The interior of the plant must be divided into zones according to the radiation levels, and suitable radiation shielding must be provided.

- Reactor building
  - The reactor building consists of the reactor containment facility and the associated systems are installed. Figure 9-1 shows a cross-section of the reactor building.

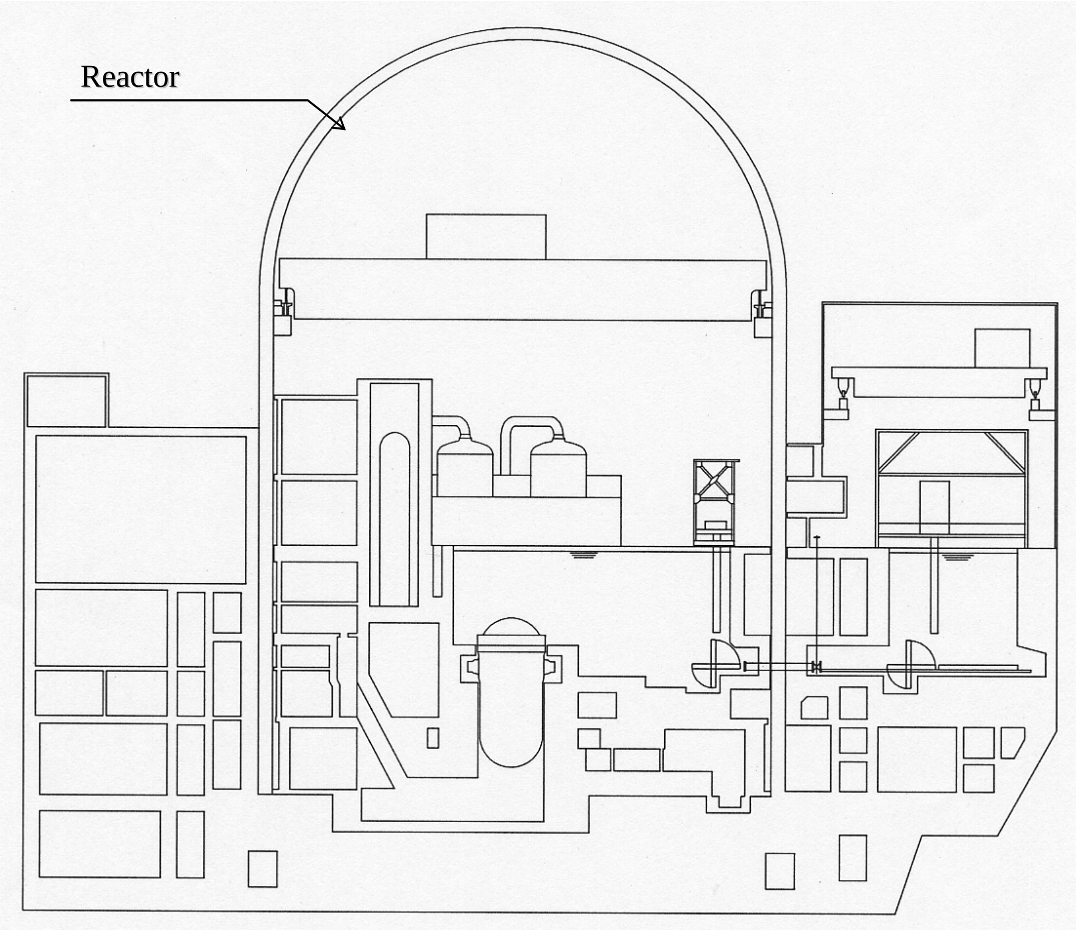

FIG. 9-1. APWR - Reactor building cross-section

- Containment
  - The containment is part of the reactor containment facility and includes the internal concrete and the annulus compartment. The reactor containment facility is also part of the engineered safety systems which include the emergency core cooling system, the containment spray system, and the annulus air purification system, etc.
  - The containment system is designed to suppress or prevent the possible dispersion of the large quantities of radioactive materials which would be released if extensive fuel failures were to occur in the reactor resulting from damage or failure of the reactor facilities such as the primary cooling system, main steam system, and feedwater system.
  - The leakage preventing function of the containment is provided by a steel liner on the inner surface while the pressure withstanding function is provided by the concrete structure. An enclosed space (annulus compartment) surrounds the lower part of the containment shell to provide a double containment and the containment penetrations for pipes, cables, ducts, and air locks pass through the annulus compartment.
  - The containment is designed so that the leak-rate is less than 0.1% per day of the weight of air in the containment at a pressure of 0.9 $\times$ maximum design air pressures at normal temperatures.
  - The containment is provided so that the general public will not be affected by radiation if it leaks at this leak-rate even if the facilities related to the primary coolant system fail or are damaged. Therefore,

severe accidents must also be carefully considred to ensure the integrity of the containment.
    - In current Japanese PWRs, the refuelling water which is the water supply used after an accident, is stored in a tank outside the containment. In an APWR, however, in order to avoid a failure to switch over the water source from the tank to the recirculation sump inside the containment, the refuelling water is stored in a pit inside the containment.
    - Also, a proper space has been provided below the reactor vessel so that debris will be distributed thinly if a hypothetical ejection of molten debris occurs, and the space is shaped to catch the debris easily to prevent it from being splashed, as far as possible, into the general spaces of the containment.

- Turbine building
    - The turbine generator, condensate and feedwater system auxiliary equipment, and other equipment are installed in the turbine building. The foundation of the turbine building is made of concrete to reduce the thickness of the mat.
    - The floor of the turbine building below ground level is made of concrete, and the floors above ground are steel structures which are designed to withstand all loads including the load of the overhead travelling crane.
    - The turbine generator systems are arranged so that the space can be utilized effectively not only during the construction of the plant but also during operation and periodical inspections.
    - Suitable spaces have been provided for inspection access, transportation of tools for inspections and maintenance, and disassembly in a way that reduces the volume of the building.

- Other buildings
    - The buildings and systems have been arranged so as to optimize the relation between the systems, separation of safety system equipment, seismic resistance, maintenance, etc.
    - Control building
      The control building which is common to both units, contains mainly the main control room, electrical equipment and access control equipment.
    - Auxiliary building
      The Auxiliary building which is common to both units, mainly houses the radioactive waste treatment systems.

10. *Plant Performance*

In the APWR, top priority is given to assuring safety and reliability, and at the same time, economic efficiency improvement is also made.

- Reduction in construction cost
    - Through the effect of economies of scale achieved by making the plant output larger by about 30% compared with the existing Japanese 4-loop plant, the unit construction cost can be reduced.
    - In addition, the construction cost is also reduced by simplifying the ECCS system, such as dividing the safety system equipment into four sub-systems, adopting a high performance accumulator tank, installation of the refueling water storage pit inside the reactor containment vessel,

as well as by reducing the amount of cables by adoption of optical multiplex transmission, adopting compact equipment, such as improved steam generators, plate heat exchangers, energy absorbing supports, making the building compact by rational arrangement of equipment through the utilization of 3D-CAD, etc.

- Shortening of construction schedule
  - Shortening of the construction schedule is being studied by adopting super heavy-duty cranes, increasing the number of large piping modules and components, etc.

- Shortening of maintenance outage
  - The regular maintenance outage can be shortened by adopting automatic and high-speed fuel handling system, bolting tensioning machine for reactor vessel opening and restoration, etc. Thus, the plant availability factor is expected to be improved.

- Reduction in exposure to radiation
  - Reduction in employees' exposure to radiation can be expected by adopting the fuel assembly zircaloy grid, optimizing pH control in RCS by application of enriched boron, enhancement of purification capacity by increasing the flow rate for purification, etc. as the measures to reduce the source of radiation.

- Project status and planned schedule
  - With the summarized technologies on PWR that have been improved based on the construction of PWRs and the experience in their operations, APWR has been improved remarkably on safety, reliability, operability, maintainability, and economic efficiency. It is expected to contribute largely to the supply of energy as the standard model of the PWRs which will be constructed in Japan including the Tsuruga-3 and 4.

11. Development status of technologies relevant to the NPP

No information is provided now.

12. Deployment status and planned schedule (1 – 2 pages)

No information is provided now.

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Appendix: **Summarized Technical Data (APWR)**

General plant data		
Reactor thermal output	4,466	MWth
Power plant output, gross	1,538	MWe
Power plant output, net		MWe
Power plant efficiency, net	34.4	%
Mode of operation		(baseload, load follow)
Plant design life	60	Years
Plant availability target		%
Seismic design, SSE		
Primary Coolant material	Light water	
Secondary Coolant material	Light water	
Moderator material	Light water	
Thermodynamic Cycle	Rankine	
Type of Cycle	Indirect	
Non-electric application		Desalination? District heat? Industrial cogeneration? H2 production?
Safety goals		
Core damage frequency		/RY
Large early release frequency		/RY
Occupational radiation exposure		Person-Sv/RY
Operator Action Time		hours
Nuclear steam supply system		
Steam flow rate at nominal conditions	8.93×10^6	kg/h
Steam pressure/temperature		MPa(a)/℃
Feedwater flow rate at nominal conditions		kg/s
Feedwater temperature		℃
Reactor coolant system		
Primary coolant flow rate	77.3×10^6	kg/h
Reactor operating pressure	15.4	MPa(a)
Core coolant inlet temperature	approximately 289	℃
Core coolant outlet temperature	approximately 325	℃
Mean temperature rise across core		℃
Reactor core		
Active core height	approximately 3.7	m
Equivalent core diameter	approximately 3.9	m
Average linear heat rate	approximately 17.6	kW/m
Average fuel power density		kW/kgU
Average core power density	approximately 103	MW/l
Fuel material	Sintered UO_2	

Cladding tube material	Zr base alloy	
Outer diameter of fuel rods	approximately 9.5	mm
Rod array of a fuel assembly	square, 17×17	
Number of fuel assemblies	257	
Enrichment of reload fuel at equilibrium core		Wt%
Fuel cycle length		months
Average discharge burnup of fuel		MWd/kg
Burnable absorber (strategy/material)		kg
Control rod absorber material	Ag-In-Cd	
Soluble neutron absorber		
Reactor pressure vessel		
Inner diameter of cylindrical shell	approximately 5 200	mm
Wall thickness of cylindrical shell		mm
Total height, inside	approximately 13,600	mm
Base material		
Design pressure/temperature		MPa(a)/℃
Transport weight		t
Steam generator		
Type	70F-1, U-tube	
Number	4	
Total tube outside surface area	approximately 6,500	m^2
Number of heat exchanger tubes	5,380	
Tube outside diameter	approximately 19	mm
Tube material	TT 690 alloy	
Transport weight		t
Reactor coolant pump		
Type	MA25S, Single-stage, centrifugal	
Number	4	
Head at rated conditions	approximately 91	m
Flow at rated conditions	25,800	m^3/h
Pump speed		rpm
Pressurizer		
Total volume		m^3
Steam volume: full power/zero power		m^3
Heating power of heater rods		kW
Primary containment		
Type	Dry,	
Overall form (spherical/cylindrical)	cylindrical, steel	
Dimensions (diameter/height)	approximately 45.5/69	m
Design pressure/temperature		MPa(a)/℃
Design leakage rate	<0.1	Vol%/day
Is secondary containment provided?		
Residual heat removal systems		
Active/passive systems		

Safety injection systems		
Active/passive systems		
Turbine		
Type of turbines	TC6F54	
Number of turbine sections per unit (e.g. HP/MP/LP)		
Turbine speed		rpm
HP turbine inlet pressure/temperature		MPa(a)/℃
Generator		
Type		
Rated power	1,715	MVA
Active power		MW
Voltage	30	kV
Frequency		Hz
Total generator mass including exciter		t
Condenser		
Type		
Condenser pressure		kPa(a)
Feedwater pumps		
Type		
Number	3	
Head at rated conditions		m
Flow at rated conditions		m^3/s
Pump speed		rpm