Boiling Water Reactor Power Plant

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Chapter 1. BWR Development

1.1. General

Boiling water reactors (BWRs) are nuclear power reactors utilizing light water as the reactor coolant and moderator to generate electricity by directly boiling the light water in a reactor core to make steam that is delivered to a turbine generator. There are two operating BWR types, roughly speaking, i.e., BWRs and ABWRs (advanced boiling water reactors)

The outline of a BWR power plant is shown in Figure 1.

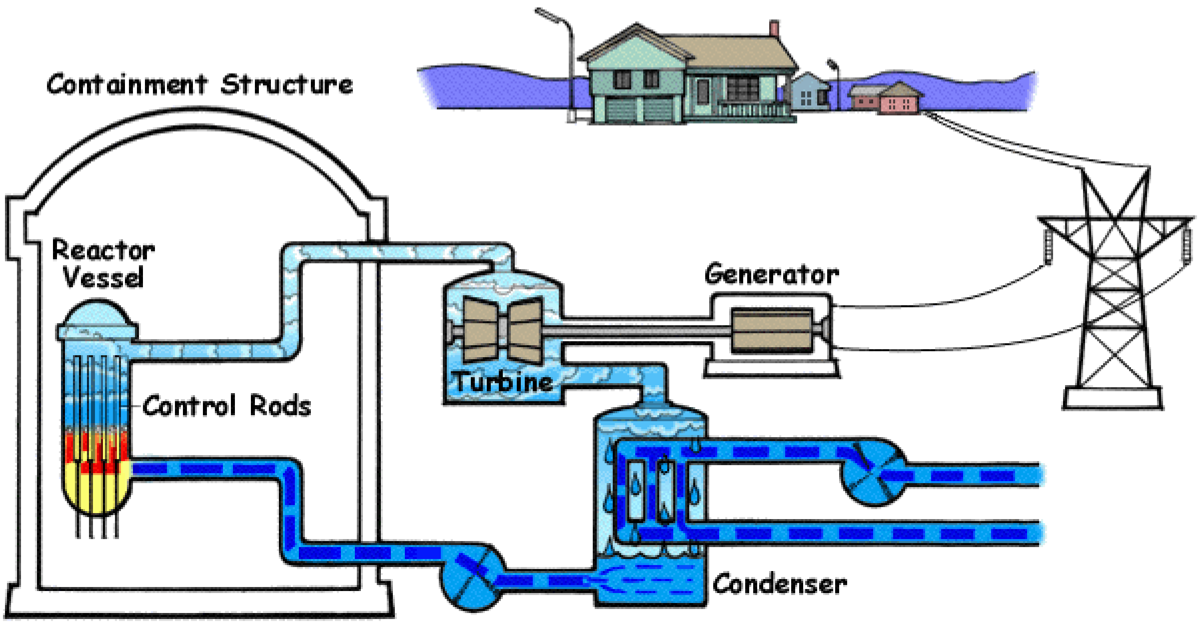


Figure 1. Outline of BWR Power Plant

More details on the System Outline of ABWR Power Plant

A pressurized water reactor (PWR) was the first type of light-water reactor developed because of its application to submarine propulsion. The civilian motivation for the BWR is reducing costs for commercial applications through design simplification and lower pressure components.

In contrast to the pressurized water reactors that utilize a primary and secondary loop, in civilian BWRs the steam going to the turbine that powers the electrical generator is produced in the reactor core rather than in steam generators or heat exchangers. There is just a single circuit in a civilian BWR in which the water is at lower pressure (about 75 times atmospheric pressure) so that it boils in the core at about 285°C.

BWRs have been originally developed by GE. GE started its development in 1950s as light water reactor type nuclear power reactors, and the Dresden Unit-1 (200,000 kWe) commissioned in July 1960 is the first BWR nuclear power station. After that, the GE company has supplied many BWRs, Siemens (KWU, Germany), ABB-Atom (Switzerland/Sweden) and Toshiba and Hitachi (Japan) also supplied many BWRS. In the following, features and types of BWRs, mainly of conventional BWRs, are explained and those of ABWRs are addressed in the next.

For BWRs, the steam void due to reactor coolant boiling has a negative-reactivity effect, which can suppress a power rise even if a positive reactivity is added. The reactor power can be controlled by two methods: reactor-coolant recirculation-flow control and control rod operation.

A BWR nuclear power plant consists of the reactor coolant recirculation system and main steam system that compose a nuclear reactor, engineered safety features that consist of the emergency core cooling system, reactor core isolation cooling system, containment cooling system and boric-acid injection system, turbine and generator equipment and other systems, such as the reactor coolant purification system, waste processing equipment, fuel handling equipment, other auxiliary equipment, etc.

1.2. BWR Type

Major reactor core parameters of BWR-2 to BWR-4, which are in operation in Japan are shown in Table 1.

Table 1 Main Parameters for BWR Core

No.	Item	Tsuruga Unit-1 (BWR-2)	Fukushima Unit-1 (BWR-3)	Hamaoka Unit-2 (BWR-4)	Tokai Unit-2 (BWR-5)	Kashiwazaki Unit-6 (ABWR)
(1)	Thermal output (MW)	1064	1380	2436	3293	3926
(2)	Electric output (MW)	357	460	840	1100	1356
(3)	Core equivalent dia. (m)	3.02	3.44	4.07	4.75	5.16
(4)	Core effective height (m)	3.66	3.66	3.71	3.71	3.71
(5)	Fuel assemblies (Number)	308	400	560	764	872
(6)	Control rod (Number)	73	97	137	185	205
(7)	Power density (kw/l)	About 40	About 40	About 50	About 50	About. 50

Improvement and history of BWR fuel in Japan are shown in Table 2. In 1960s, the development started including introduction of overseas technologies under license agreements, and the fuel type has been changed from 6x6 to 9x9 adopting many improvements resulting from nuclear and mechanical research and developments.

Table 2 BWR Fuel Improvement in Japan

Year/Item	Objective	Major Improvement	Fuel Type	Reactor Type
1960	Development in general	Basic study on fuel material Fuel rod irradiation test Core design study Fuel manufacturing technologies	6x6 7x7	JPDR (BWR-1)
1970	Initial performance development	6x6 type fuel demonstration Domestic fuel performance demonstration 7x7 type fuel development (high power density and long fuel rod development)	7x7R	Tsuruga-1 (BWR-2) Fukushima I-1 (BWR-3)
1980	Reliability improvement	Reliability improvement Improved 7x7 type fuel development Preconditioning fuel operation 8x8 type fuel development	8x8 8x8 R	Fukushima I-2 (BWR-4) Tokai-2 (BWR-5)
1990	Availability improvement	Re-evaluation of preconditioning fuel operation He pressurized fuel Two regional fuel reactivity design Controlled cell core	Liner 8x8 R High Burnup 8x8	Fukushima II-2 (Improved BWR-5)
2000	High performance / high burnup fuel	Zirconium liner fuel High burnup fuel		Kashiwazaki-6 (ABWR)

Improvement of BWR containment is shown in Figure 2. Five types of containments were applied for Japanese BWRs. Typical design for each type of containment is illustrated with major dimensions. The design has attained significant improvement in the total volume per output, resulting in a large cost benefit.

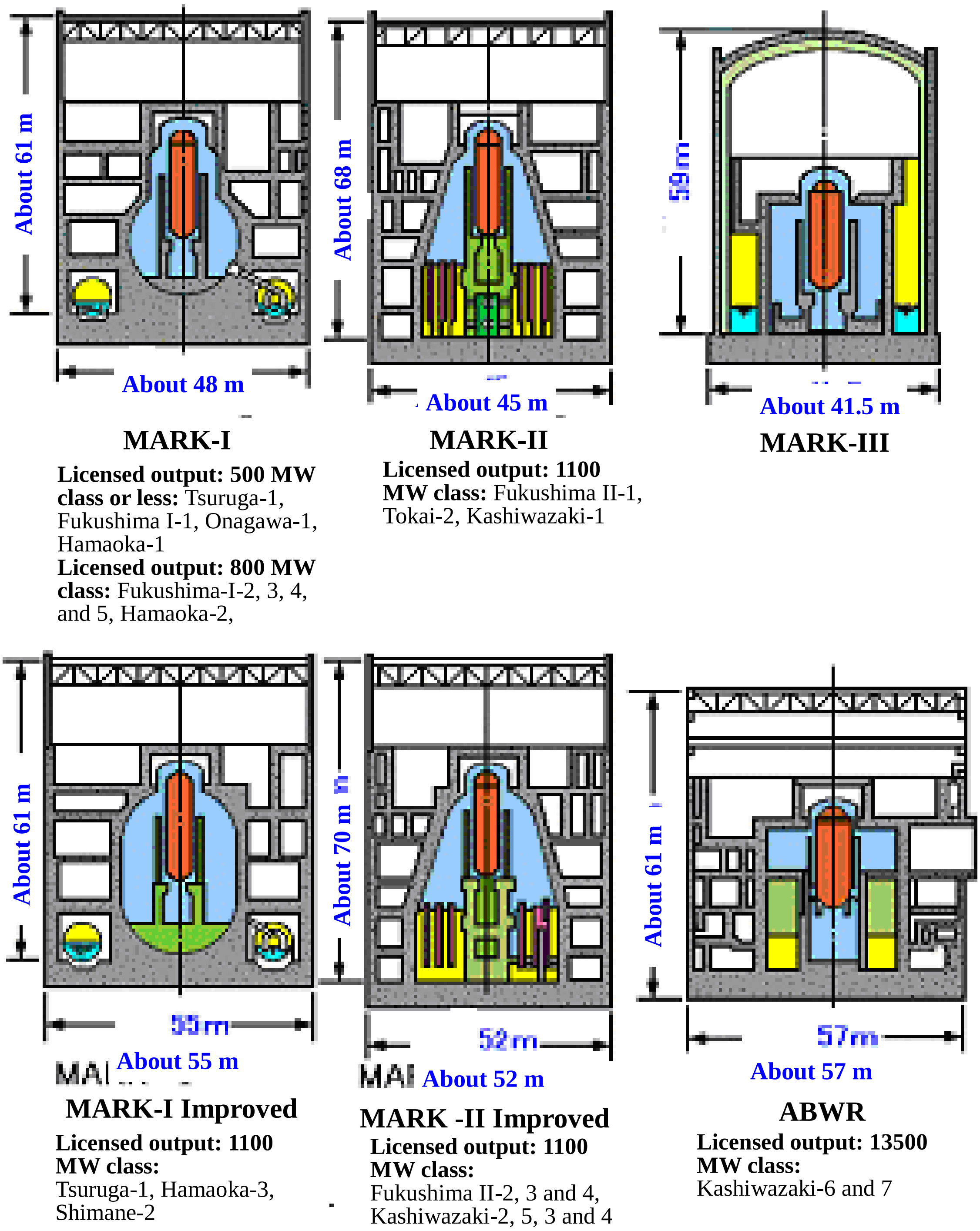


Figure 2. History of BWR Containment

There are two operating BWR types, roughly speaking, i.e. BWRs including their modifications and ABWRs (advanced BWRs). The first commercial power reactor constructed in U.S. was the Dresden Unit-1 (full power operation in July 1960), which was the BWR-1 reactor. This BWR-1 reactor was dual cycle like a pressurized water reactor and adopted a dry type reactor containment vessel. The BWR-2 and the subsequent ones ware designed to increase the power density that results in a smaller core size, to simplify the system adopting a direct cycle with a steam drum provided inside a reactor vessel, to multiplex the emergency core cooling system (ECCS), and to reduce the containment vessel volume adopting a pressure-suppression-type pool, which led to the current operating BWR designs.

Chapter 2. BWR Technologies

2.1. Reactor Coolant Recirculation System and Main Steam System

Boiling water reactors (BWRs) are nuclear power reactors generating electricity by directly boiling the light water in a reactor pressure vessel to make steam that is delivered to a turbine generator. After driving a turbine, the steam is converted into water with a condenser (cooled by sea water in Japan), and pumped into the reactor vessel with feedwater pumps. A part of the water is sent into the reactor vessel after being pressurized with recirculation pumps installed outside of the vessel and fed into the reactor core from the bottom part of the reactor vessel with jet pumps.

Inside of a BWR reactor pressure vessel (RPV), feedwater enters through nozzles high on the vessel, well above the top of the nuclear fuel assemblies (these nuclear fuel assemblies constitute the "core") but below the water level. The feedwater is pumped into the RPV from the condensers located underneath the low pressure turbines and after going through feedwater heaters that raise its temperature using extraction steam from various turbine stages.

The feedwater enters into the downcomer region and combines with water exiting the water separators. The feedwater subcools the saturated water from the steam separators. This water now flows down the downcomer region, which is separated from the core by a tall shroud. The water then goes through either jet pumps or reactor internal pumps that provide additional pumping power (hydraulic head). The water now makes a 180 degree turn and moves up through the lower core plate into the nuclear core where the fuel elements heat the water. When the flow moves out of the core through the upper core plate, about 12–15% of the volume of the flow is saturated steam.

2.2. Structure of BWRs

(1) BWR reactor core and internals

Reactor core and internal structures of 1,100MWe class BWR reactor vessel are shown in Figure 3. In a reactor vessel, there are a reactor core that mainly consists of fuel assemblies and control rods in the center, equipment for generating steam for a turbine, such as a steam-water separator and a steam dryer in the upper part of the vessel, equipment for reactor-power control, such as control rod guide tubes and control rod drive housings in the lower part of the vessel, and a core shroud, jet pumps etc. that surrounds the reactor core and composes the coolant flow path in the periphery of reactor core.

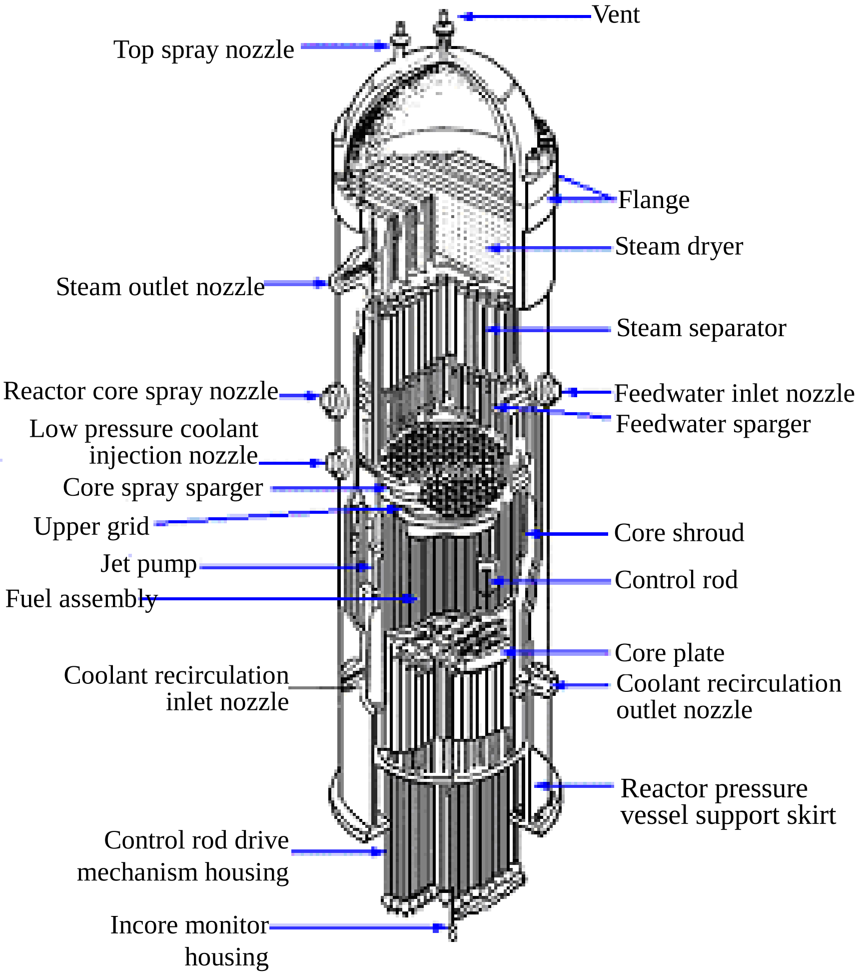


Figure 3. Internal Structure of BWR Reactor Vessel

(2) Nuclear fuel

BWR fuel assemblies, for an example of 8x8 type, consists of 64 rods: 62 fuel rods, one spacer holding water rod and one water rod, which are arranged to a tetragonal lattice of 8x8 and enclosed in a channel box made of zircaloy as shown in Figure 4. Fuel rods are structured to contain uranium-dioxide pellets, a plenum spring etc. in a zircaloy cladding tube, of which both ends are weld-sealed with end plugs after pressurized with helium gas. The plenum is a space provided so that the fission gas discharged from fuel pellets accompanying fuel burnup is accommodated and the fuel rod internal pressure does not become excessive.

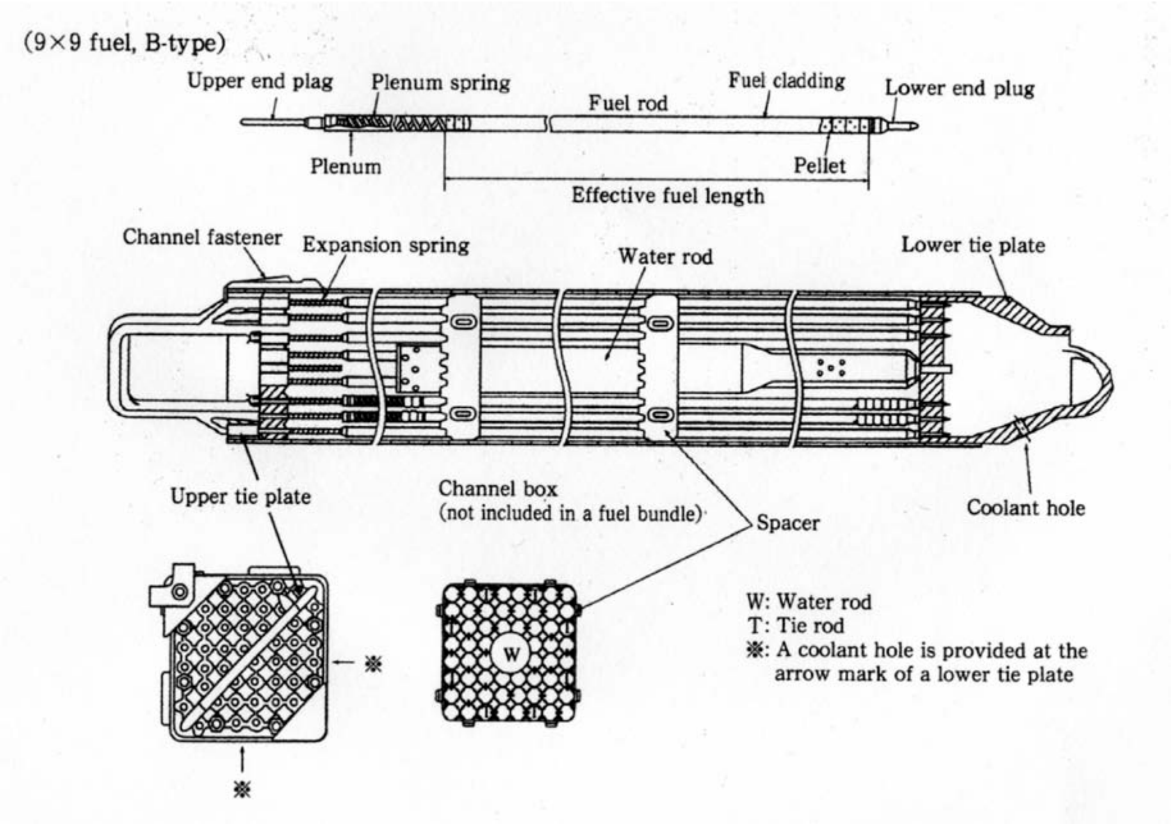


Figure 4. **BWR Nuclear Fuel Structure**

(3) Control rod and its drive mechanism

BWR control rods are composed of blades in a shape of cruciform in order to move through the gaps formed between four channels of fuel assemblies as shown in Figure 5. Types of control rods are, in terms of the absorber materials, boron carbide (B_4C), hafnium (Hf) and combination of these. A velocity limiter of an umbrella shape is provided at the lower portion of the control rod to slow down the dropping velocity in case of a control rod drop accident. Moreover, a connector to couple a control rod to a control rod drive mechanism is provided.

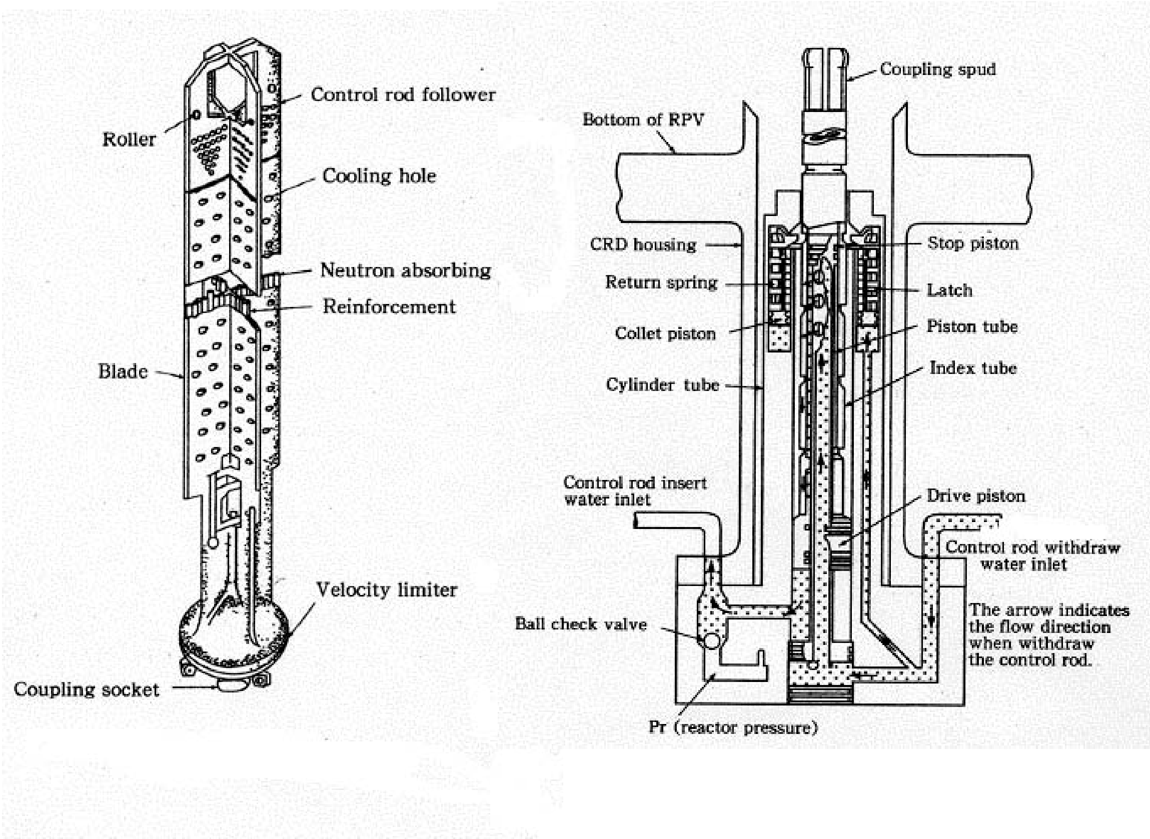


Figure 5. BWR Control Rod and its Drive Mechanism

There are two types of the control rod drive mechanism: hydraulic pressure drive and motor drive. Both types utilize the nitrogen-gas pressure stored in accumulators as driving power for fast insertion of control rods. When an anomaly occurs or could occur at a nuclear reactor, the fast insertion of all control rods into a reactor core is carried out all at once from the lower part of reactor core to shutdown nuclear reactor operation (it is called that a nuclear reactor is scrammed.) The boric acid solution injection system is provided to inject a neutron absorber material into the reactor core to stop reactor operation when the control rods cannot be inserted and the nuclear reactor cannot be placed in low-temperature shutdown mode.

2.3. Engineered Safety Feature

(1) Emergency core cooling system

At an abnormal event of a BWR, actuation of the reactor shutdown system (a part of the safety protection system) stops the nuclear reactor operation securely. The emergency core cooling system (ECCS) is provided for the case when a break accident occurs to reactor coolant system piping etc. and the reactor coolant is lost from a reactor core (loss of coolant accident, LOCA). This system consists of one high pressure core cooling system, one low pressure core cooling system, and three low pressure core injection (reflooder) systems.

More Details on Safety Design

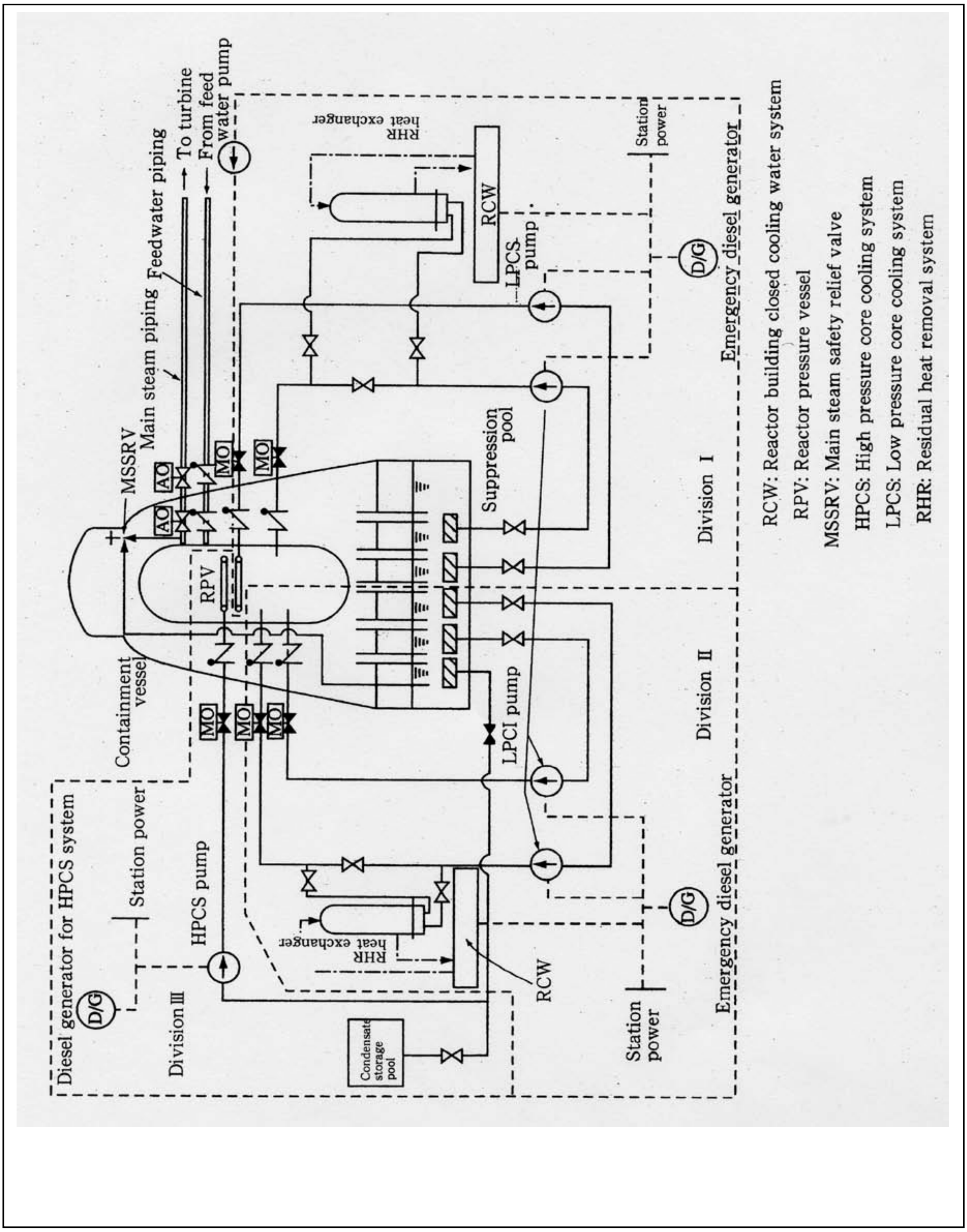


Figure 6. ECCS Network for BWR-5, 1100MWe

(2) Reactor containment

Radioactive materials are released into the high temperature and high pressure coolant when a fuel failure occurs. Therefore, a reactor containment is provided so that the coolant would not discharge to the outside (Figure 7). All BWR containments are pressure suppression (pressure suppression pool) type, and the steam discharged into the containment is led to the water pool of the pressure suppression chamber, cooled and condensed, and the pressure rise within the containment is suppressed as a result. Moreover, as the temperature and pressure of the containment rise due to the fuel decay heat in a long term after an accident, it is necessary to cool the inside of the containment. Furthermore, it is also necessary to remove radioactive materials such as iodine within the containment. For such purposes, the containment spray system is provided within the containment (drywell spray, pressure suppression chamber spray). Furthermore, the standby gas treatment system is provided in the reactor building so that the radioactive materials will not be released to the outside of the containment.

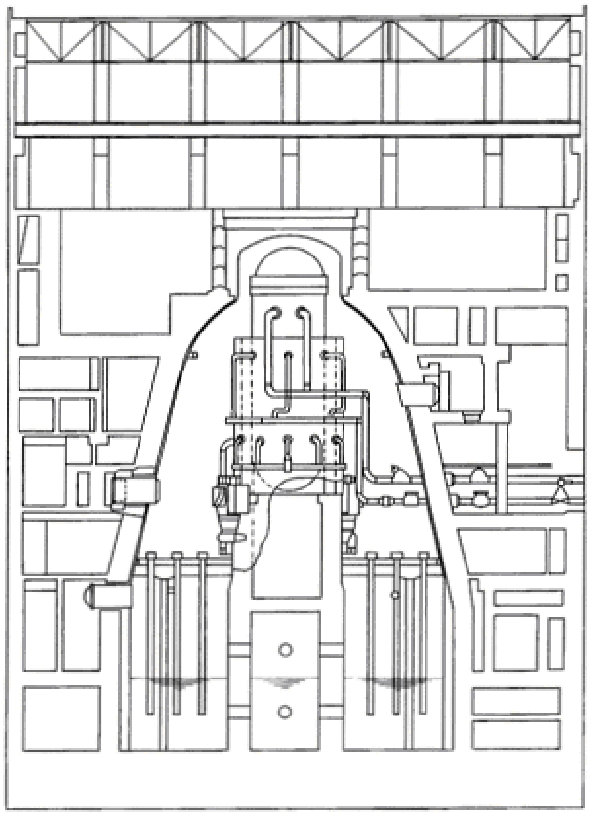


Figure 7. BWR Containment in the Reactor Building (Improved Mark-II)

In addition, following a loss of coolant accident, the temperature of fuel cladding could rise and hydrogen could be generated by a water-metal reaction, which could impair the containment integrity due to hydrogen gas combustion. In order to prevent such a case, BWR containments are kept inert with nitrogen gas (Mark-III type containment is designed not to use the nitrogen gas, but it is not adopted in Japan) during normal operation, and the flammability control system to prevent hydrogen combustion by recombining the generated hydrogen gas with oxygen gas.

2.4. Other Systems and Equipment

(1) Reactor coolant clean up system

The reactor coolant clean up system is provided to keep the coolant purity high, and consists of pumps, regenerative heat exchangers, non-regenerative heat exchangers, filter demineralizers, auxiliary equipment, etc.

The reactor coolant clean up system, together with the condensate cleanup system, keeps the coolant properties within the following values;

Electric conductivity (25 degrees C)	1 micro-S / cm or less
Cr	0.1 ppm or less
pH (25 degrees C)	5.6 - 8.6

(2) Reactor core isolation cooling system

The reactor core isolation cooling system is provided to inject the condensed water of residual heat removal system or condensate storage tank water, etc. into a reactor core with the turbine-driven pump using a part of the nuclear reactor steam to maintain the reactor water level, when supply of the condensate or feed water is stopped due to a certain cause after the reactor shutdown.

(3) Residual heat removal system

The residual heat removal system is provided for removal of the residual heat during a normal reactor shutdown and nuclear reactor isolation condition and for core cooling in case of a loss of coolant accident, etc.

The system consists of three independent loops, consisting of two sets of heat exchangers and three sets of pumps, which can be used in four modes by changing valve lineup. In addition, the system can cool the fuel pool using a connection line to the fuel pool cooling and cleanup system, when required.

(4) Waste processing system

Wastes generated in a plant are divided into gas, liquid and solid materials, and are processed separately. The gaseous waste, after attenuating the radioactivity to sufficiently low level with an activated-carbon-type noble gas hold-up device, is discharged from a vent stack monitoring the concentrations of radioactive materials. The liquid waste, after being collected from each generating source, is processed with a filter, a demineralizer and a waste evaporator, and is reused as make-up water or discharged. The liquid waste condensed with the waste evaporator is processed as a solid waste. The solid waste is processed by solidification, incineration, compression etc. corresponding to the type and canned in a drum for storage in a storage facility. In the solidification method, there are bituminization, plastic solidification and cement solidification.

(5) Fuel handling equipment

Refueling is carried out once per 12 to 24 months in principle for an equilibrium cycle, and the required refueling time period is about 20 days. The number of removed fuel assemblies at one refueling is 20 to 30% of the total fuel assemblies in a core.

(6) Fuel pool cooling and cleanup system

The fuel pool cooling and cleanup system is provided to remove the decay heat of the spent fuel with the heat exchangers of the reactor building closed cooling water system to cool the fuel pool water, and to maintain the water purity and visibility of the fuel pool, reactor well and pit for the steam dryer and steam-water separator by filter-demineralization of the fuel pool water with a filter demineralizer,

The fuel pool cooling and cleanup system consists of pumps, filter demineralizers, heat exchangers, auxiliary equipment etc.

(7) Turbine-generator equipment

(a) Steam turbine

Generally speaking, the steam turbine for nuclear power consumes more steam per unit output and is a larger size compared with the turbine for thermal power plants, as the turbine inlet steam condition is not good compared with that for thermal power plants.

Therefore, the rotation frequency of both the high-pressure and low pressure turbines is 1,500 to 1,800 rpm.

(b) Generator

The turbine generator for nuclear power plants has no essential difference from that for thermal power plants.

2.5. Power Control of BWR

(1) Power control method and self-regulating characteristics

The BWR generates steam with pressure about 70 kg/cm^2 by boiling light water in the reactor core. Moreover, the amount of steam bubbles (void) generated by the boiling is controlled with recirculation pumps (variable velocity pump) to control the nuclear reaction (power), which is called the recirculation flow control system. As control rods are withdrawn out of the core, the reactivity increases and then, the power (heat generation) increases, which results in increase of steam void leading to reduction of moderator density, and the rate of uranium fission becomes small and the reactivity decreases, which balances and stabilizes the reactor power (reactivity). As control rods are inserted into the core, the reactivity decreases and the power decreases, which results in decrease of steam void leading to increase of moderator density, and the rate of uranium fission becomes large and the reactivity increases, which balances and stabilizes the reactor power. In this way, BWRs have a self-regulating characteristic of the reactor power.

(2) Heat transfer and power control

The heat generated in fuel rods is transferred to the reactor coolant. The magnitude of heat transferred according to the temperature difference between the heat transfer surface and the coolant has been obtained in many experiments. Since the heat transfer decreases in the transition film-boiling region in which the boiling becomes violent that could cause a burnout of fuel cladding tube, the heat transfer in the nucleate-boiling region is utilized in BWR. Therefore, the reactor operation limits are imposed on BWRs not to approach to the transition film-boiling region during normal operation and abnormal operational transients.

(3) Load fluctuation and reactor pressure reduction

When BWRs experience a load fluctuation in automatic power control mode, first of all, the reactor power is adjusted by increase or decrease of the recirculation flow. Automatic power control is adjusted during about 70%--100% of the rated power. If electrical grid demands increase turbine generator output power, at first the power control system increases the recirculation flow that results in increase of the reactor power. The reactor pressure is controlled to be constant by opening of a turbine control valve by reactor pressure system. Opening of a turbine control valve increases the steam flow and the turbine generator output power. This method is called "the reactor master / turbine slave (nuclear reactor priority method)." In addition, when an abnormal turbine trip occurs, the steam flow is interrupted and the reactor scram occurs to protect abnormal pressure rise. Also, bypass valves are opened to bypass the steam to main condenser.

Chapter 3. Features of BWR

The BWR is characterized by two-phase fluid flow (water and steam) in the upper part of the reactor core. Light water (i.e., common distilled water) is the working fluid used to conduct heat away from the nuclear fuel. The water around the fuel elements also "thermalizes" neutrons, i.e., reduces their kinetic energy, which is necessary to improve the probability of fission of fissile fuel. Fissile fuel material, such as the U-235 and Pu-239 isotopes, have large capture cross sections for thermal neutrons.

3.1. BWR Design

(1) Generation of steam in a reactor core

In contrast to the pressurized water reactors that utilize a primary and secondary loop, in civilian BWRs the steam going to the turbine that powers the electrical generator is produced in the reactor core rather than in steam generators or heat exchangers. There is just a single circuit in a civilian BWR in which the water is at lower pressure (about 75 times atmospheric pressure) compared to a PWR so that it boils in the core at about 285°C. The reactor is designed to operate with steam comprising 12 to 15% of the volume of the two-phase coolant flow (the "void fraction") in the top part of the core, resulting in less moderation, lower neutron efficiency and lower power density than in the bottom part of the core. In comparison, there is no significant boiling allowed in a PWR because of the high pressure maintained in its primary loop (about 158 times atmospheric pressure).

(2) Feed water system

Inside of a BWR reactor pressure vessel (RPV), feedwater enters through nozzles high on the vessel, well above the top of the nuclear fuel assemblies (these nuclear fuel assemblies constitute the "core") but below the water level. The feedwater is pumped into the RPV from the condensers located underneath the low pressure turbines and after going through feedwater heaters that raise its temperature using extraction steam from various turbine stages.

(3) Fluid recirculation in the reactor vessel

The heating from the core creates a thermal head that assists the recirculation pumps in recirculating the water inside of the RPV. A BWR can be designed with no recirculation pumps and rely entirely on the thermal head to recirculate the water inside of the RPV. The forced recirculation head from the recirculation pumps is very useful in controlling power, however. The thermal power level is easily varied by simply increasing or decreasing the speed of the recirculation pumps.

The two phase fluid (water and steam) above the core enters the riser area, which is the upper region contained inside of the shroud. The height of this region may be increased to increase the thermal natural recirculation pumping head. At the top of the riser area is the water separator. By swirling the two phase flow in cyclone separators, the steam is separated and rises upwards towards the steam dryer while the water remains behind and flows horizontally out into the downcomer region. In the downcomer region, it combines with the feedwater flow and the cycle repeats.

The saturated steam that rises above the separator is dried by a chevron dryer structure. The steam then exits the RPV through four main steam lines and goes to the turbine.

(4) Reactor power control system

Reactor power is controlled via two methods: by inserting or withdrawing control rods and by changing the water flow through the reactor core.

Positioning (withdrawing or inserting) control rods is the normal method for controlling power when starting up a BWR. As control rods are withdrawn, neutron absorption decreases in the control material and increases in the fuel, so reactor power increases. As control rods are inserted, neutron absorption increases in the control material and decreases in the fuel, so reactor power decreases. Some early BWRs and the proposed ESBWR designs use only natural circulation with control rod positioning to control power from zero to 100% because they do not have reactor recirculation systems.

Changing (increasing or decreasing) the flow of water through the core is the normal and convenient method for controlling power. When operating on the so-called "100% rod line," power may be varied from approximately 70% to 100% of rated power by changing the reactor recirculation flow by varying the speed of the recirculation pumps. As flow of water through the core is increased, steam bubbles ("voids") are more quickly removed from the core, the amount of liquid water in the core increases, neutron moderation increases, more neutrons are slowed down to be absorbed by the fuel, and reactor power increases. As flow of water through the core is decreased, steam voids remain longer in the core, the amount of liquid water in the core decreases, neutron moderation decreases, fewer neutrons are slowed down to be absorbed by the fuel, and reactor power decreases.

(5) Steam turbines

Steam produced in the reactor core passes through steam separators and dryer plates above the core and then directly to the turbine, which is part of the reactor circuit. Because the water around the core of a reactor is always contaminated with traces of radionuclides, the turbine must be shielded during normal operation, and radiological protection must be provided during maintenance. Most of the radioactivity in the water is very short-lived (mostly N-16, with a 7 second half life), so the turbine hall can be entered soon after the reactor is shut down.

(6) Size of reactor core

A modern BWR fuel assembly comprises 74 to 100 fuel rods, and there are up to approximately 800 assemblies in a reactor core, holding up to approximately 140 tonnes of uranium. The number of fuel assemblies in a specific reactor is based on considerations of desired reactor power output, reactor core size and reactor power density.

Part 2. Advanced BWRs

Chapter 4. ABWR Development

ABWRs are Generation III reactors based on the boiling water reactor. The ABWR was designed by General Electric and Japanese BWR suppliers. The standard ABWR plant design has a net output of about 1350 megawatts electrical.

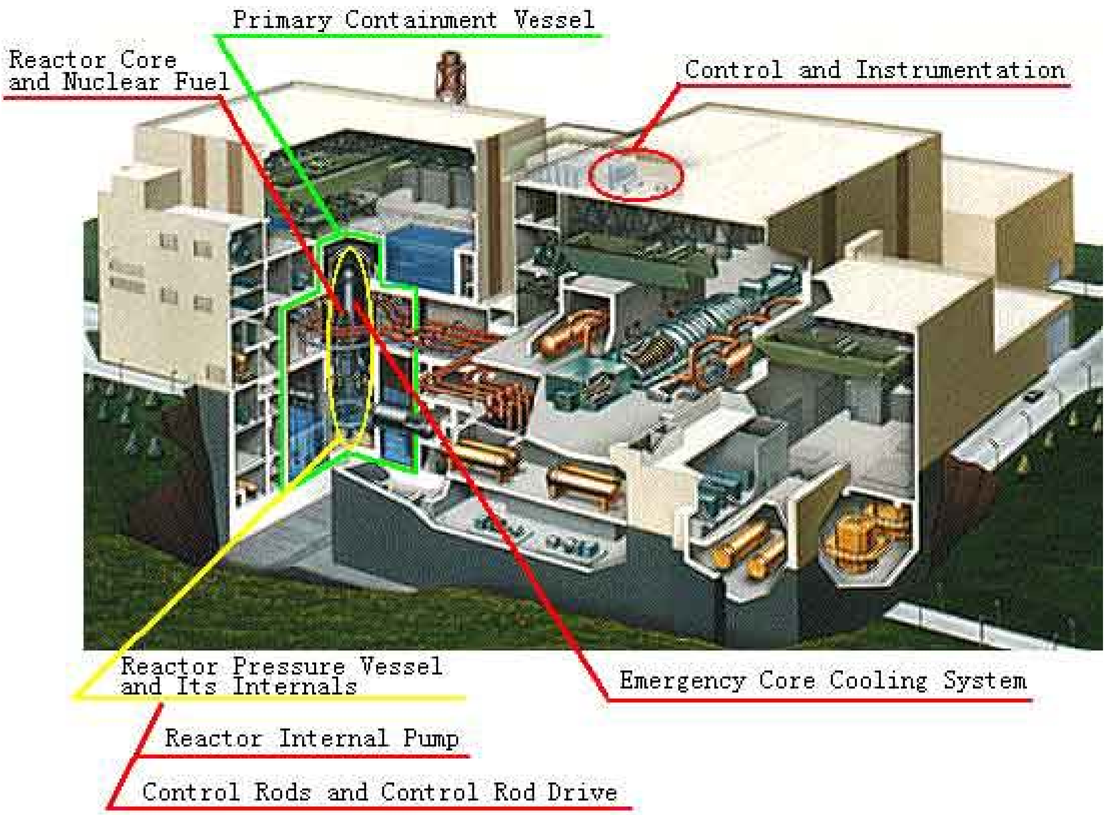


Figure 8. **ABWR Power Plant Structure**

Major differences between the BWR and ABWR designs are as shown in Table 3: the reactor coolant pump is changed from the combination of recirculation pumps and jet pumps to internal pumps (in-reactor-vessel type pump), the control rod drive system is changed to a combination of a motor-driven drive and a hydraulic pressure drive from the hydraulic pressure drive, and the containment is a reinforced-concrete type containment vessel. In addition, the kashiwazaki kariwa Unit-6 and Unit-7 (electrical output is 1,356,000kW gross, respectively) in Japan have started commercial operation as the first operating ABWRs in the world.

Table 3. Major Specifications for BWR and ABWR

Items		ABWR	Conventional BWR
Electricity output	MWe	1350 class	1100 class
Thermal output	MWt	3926	3293
Reactor pressure	kgf/cm^2g	72.1	70.7
Feed water temperature	Degree-C	215	215
Core flow	Kg/h	About $52x10^6$	About $48x10^6$
Fuel type		New-type 8x8	New-type 8x8
Number of fuel assemblies		872	764
Number of control rods		205	185
Reactor pressure vessel	ID: m H: m	About 7.1 About 21	About 6.4 About 22
Reactor water recirculation system		Reactor internal pumps (10)	Outer recirculation pumps (2) + jet pumps (20)
Control rod drive mechanism			
Power control		Fine motion CR drive (FMCRD) system	Hydraulic pressure CR drive (CRD) system
Scram		Fast scram with hydraulic pressure drive	Fast scram with hydraulic pressure drive
Steam flow restrictor		Reactor pressure vessel nozzle	Main steam pipe Venturi nozzle
Emergency core cooling system		Low pressure reflooder system (3 systems)	Low pressure reflooder system (3 systems)
		High pressure core reflooder system (2 systems)	Low pressure core spray system
		Reactor core isolation cooling system	High pressure core spray system
		Automatic depressurization system	Automatic depressurization system
Residual heat removal system		3 systems (common use)	2 systems (common use)
Containment		Building integral-type made of reinforced concrete	Advanced Mark-I or advanced Mark-II made of steel
Main turbine			
Type		TC8F52”	TC8F41”/43”
Thermal cycle		2 stage reheating	Non-reheating
Number of steam extraction stages		6	6

More details in Standard ABWR Technical Data

Following the Kashiwazaki-Kariwas Unit-6 and Unit-7, the Hamaoka Unit-5 of the Chubu Electric Power Co., Inc., which is the second generation ABWR adopting new technologies, started its commercial operation in January 2005 as the world's largest class output power station.

Chapter 5. ABWR Technologies

5.1 Features of ABWR

BWR characterized by the simplified direct cycle type is completed as a high reliability and safety nuclear reactor with many improvements, such as optimization of the core power density and fuel burnup, adoption of a built-in steam-water separator, multiple emergency core cooling system, etc. In addition to those improvements, ABWRs adopt the following superior technologies.

(1) Reactor pressure vessel and internals

The nuclear reactor of advanced building water reactor (ABWR) adopts the internal-pump system as a reactor-coolant recirculation system, which installs pumps in a reactor pressure vessel. The reactor internals consist of internal structures, such as steam-water separator and steam dryer, and a core support for fuel assemblies as shown in Figure 9.

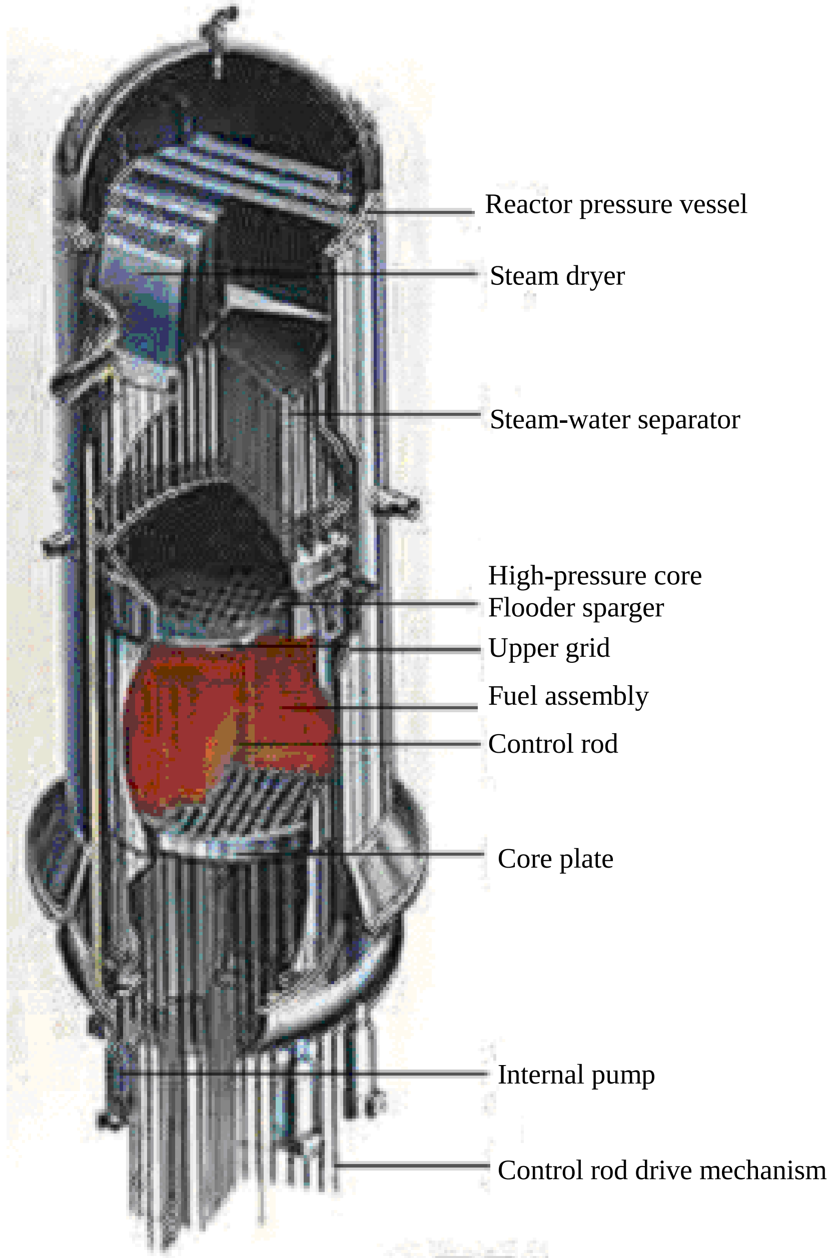


Figure 9. Reactor Pressure Vessel and Internals

Utilizing their 30 years of experience in operating BWR reactors, a special care is made in selecting the right material. Cobalt has been eliminated from the design. The steel used in the primary system is made of nuclear grade material (low carbon alloys) which are resistant to integranular stress corrosion cracking.

The ABWR reactor pressure vessel is 21 meters high and 7.1 meters in diameter.

The base metal of the reactor pressure vessel, which contains fuel assemblies, control rods and reactor internals, is made of low alloy steel and the inside surface of the vessel is lined with stainless steel to have a corrosion resistance.

Much of the vessel, including the 4 vessel rings from the core beltline to the bottom head, is made from single forging. The vessel has no nozzles greater than 2 inches in diameter anywhere below the top of the core because the external recirculation loops have been eliminated. Because of these two features, over 50% of the welds and all of the piping and pipe supports in the primary system have been eliminated and, along with it, the biggest source of occupational exposure in the BWR.

The reactor core comprises fuel assemblies as shown in Figure 10 and control rods. Each fuel rod in fuel assemblies contains sintered pellets of low-enriched uranium within a zirconium-lined cladding. They are brought together in fuel assemblies, 8x8 arrays of fuel rods held in place by upper and lower tie plates and spacers.

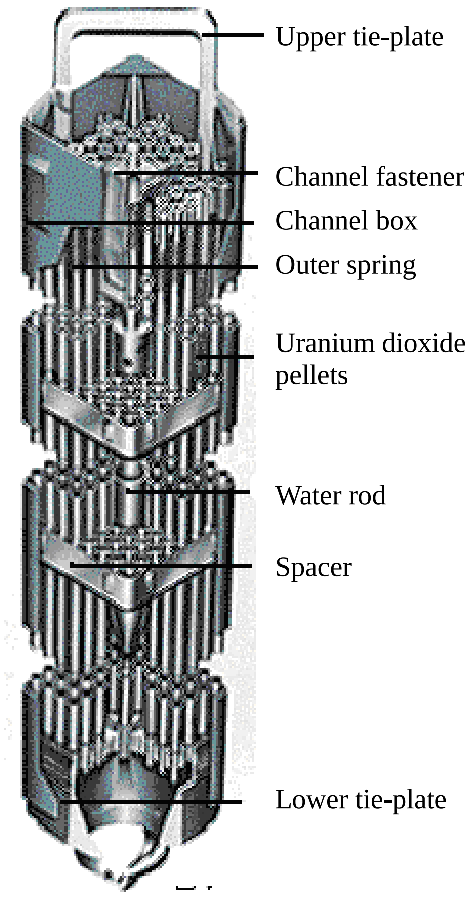


Figure 10 ABWR Fuel

(2) External recirculation system eliminated

One of the unique features of the ABWR is its external recirculation system elimination. The external recirculation pumps and piping have been replaced by ten reactor internal pumps mounted to the bottom head. (Refer to Figure 11)

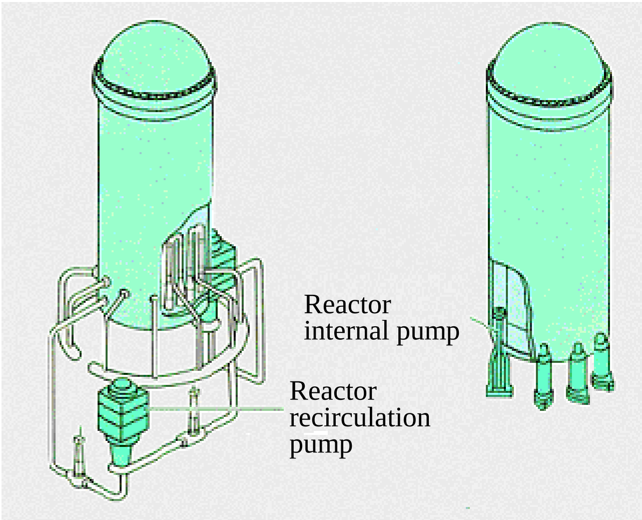


Figure 11. Reactor Cooling Pump for BWR and ABWR

Prior to the ABWR, all large commercial nuclear steam supply systems provided by GE from the BWR/3 through the BWR/6 designs used jet pump recirculation systems. These systems have two large recirculation pumps (each up to 9000 Hp) located outside of the reactor pressure vessel (RPV). Each pump takes a suction from the bottom of the downcomer region through a large diameter nozzle and discharges through multiple jet pumps inside of the RPV in the downcomer region. There is one nozzle per jet pump for the discharge back into the RPV and the external headers supplying these nozzles. Valves are required to isolate this piping in the event of a failure.

Consequently, reactor internal pumps eliminate all of the jet pumps (typically 10), all of the external piping, the isolation valves and the large diameter nozzles that penetrated the RPV.

(3) Internal pump

Reactor internal pumps inside of the reactor pressure vessel (RPV) are a major improvement over previous BWR reactor plant designs (BWR/6 and prior). These pumps are powered by wet-rotor motors with the housings connected to the bottom of the RPV and eliminating large diameter external recirculation pipes that are possible leakage paths. The 10 internal pumps are located at the bottom of the downcomer region.

The first reactors to use reactor internal pumps were designed by ASEA-Atom (now Westinghouse Electric Company by way of mergers and buyouts, which is owned by Toshiba) and built in Sweden. These plants have operated very successfully for many years.

The internal pumps reduce the required pumping power for the same flow to about half that required with the jet pump system with external recirculation loops. Thus, in addition to the safety and cost improvements due to eliminating the piping, the overall plant thermal efficiency is increased. Eliminating the external recirculation piping also reduces occupational radiation exposure to personnel during maintenance.

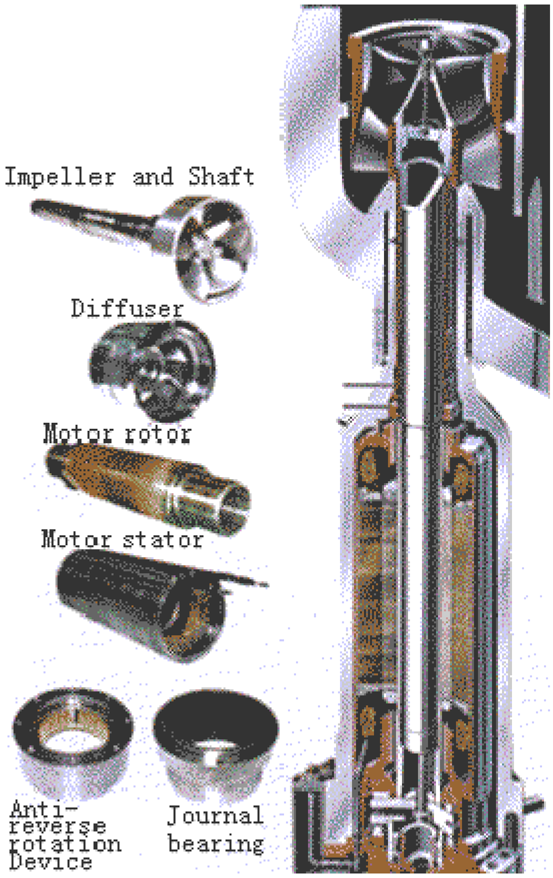


Figure 12. Reactor Internal Pump

(4) Control rod and drive mechanism

A operational feature in the ABWR design is electric fine motion control rod drives. BWRs use a hydraulic system to move the control rods which is driven by locking piston drive mechanism.

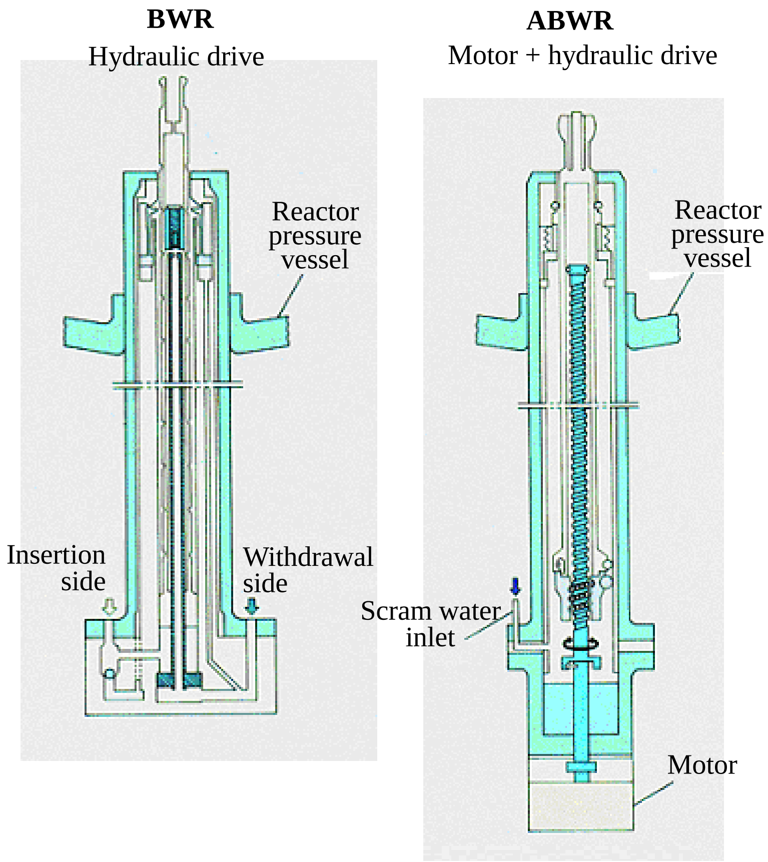


Figure 13. Control Drive Mechanism for BWR and ABWR

The materials in the control rods absorb neutrons and so restrain and control the reactor's nuclear fission chain reaction. The rods themselves have a cruciform cross section. They are inserted upwards, from the base of the RPV, into the rod spaces in fuel assemblies.

Fine motion control rod drives (FMCRD) are introduced in the ABWR. The control rods are scrammed hydraulically but can also scrammed by the electric motor as a backup. The FMCRDs have continuous clean water purge to keep radiation to very low levels.

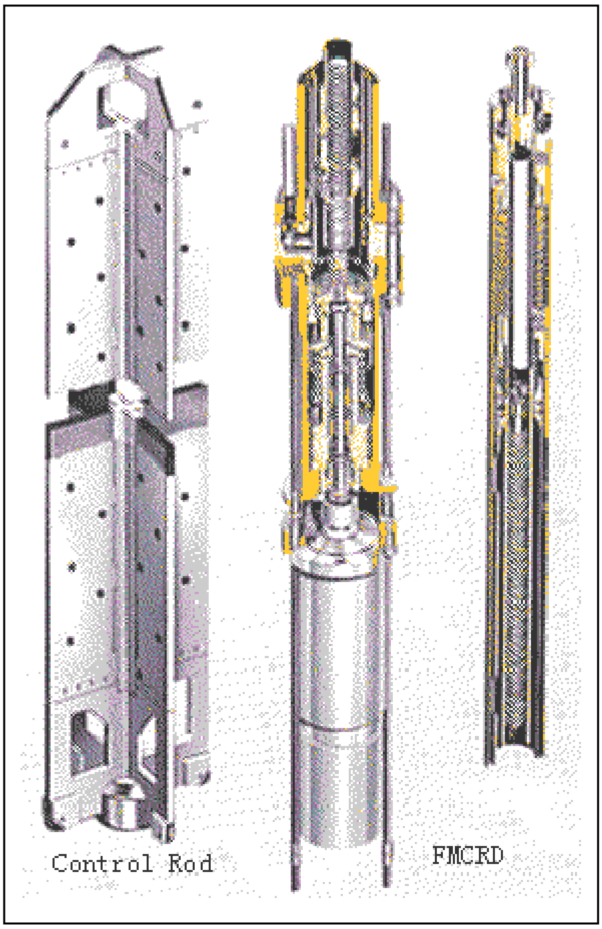


Figure 14. Control Rod and Drive Mechanism

(5) Safety - Simplified active safety systems

ABWR has three completely independent and redundant divisions of safety systems. The systems are mechanically separated and have no cross connections as in earlier BWRs. They are electronically separated so that each division has access to redundant sources of ac power and, for added safety, its own dedicated emergency diesel generator. Divisions are physically separated. Each division is located in a different quadrant of the reactor building, separated by fire walls. A fire, flood or loss of power which disables one division has no effect on the capability of the other safety systems. Finally, each division contains both a high and low pressure system and each system has its own dedicated heat exchanger to control core cooling and remove decay heat. One of the high pressure systems, the reactor core isolation cooling (RCIC) system, is powered by reactor steam and provides the diverse protection needed should there be a station blackout.

The safety systems have the capability to keep the core covered at all times. Because of this capability and the generous thermal margins built into the fuel designs, the frequency of transients which will lead to a scram and therefore to plant shutdown have been greatly reduced (to less than one per year). In the event of a loss of coolant accident, plant response has been fully automated.

Any accident resulting in a loss of reactor coolant automatically sets off the emergency core cooling system (ECCS). Made up of multiple safety systems, each one functioning independently, ECCS also has its own diesel-driven standby generators that take over if external power is lost.

High pressure core flooder (HPCF) and reactor core isolation cooling (RCIC) systems: These systems inject water into the core to cool it and reduce reactor pressure.

Low pressure flooder (LPFL) system**:** Once pressure in the reactor vessel is reduced, this system injects water into the reactor vessel. The reactor core is then cooled safely.

Automatic-depressurization system: Should the high-pressure injection system fails, this system lowers the reactor vessel pressure to a level where the LPFL system can function.

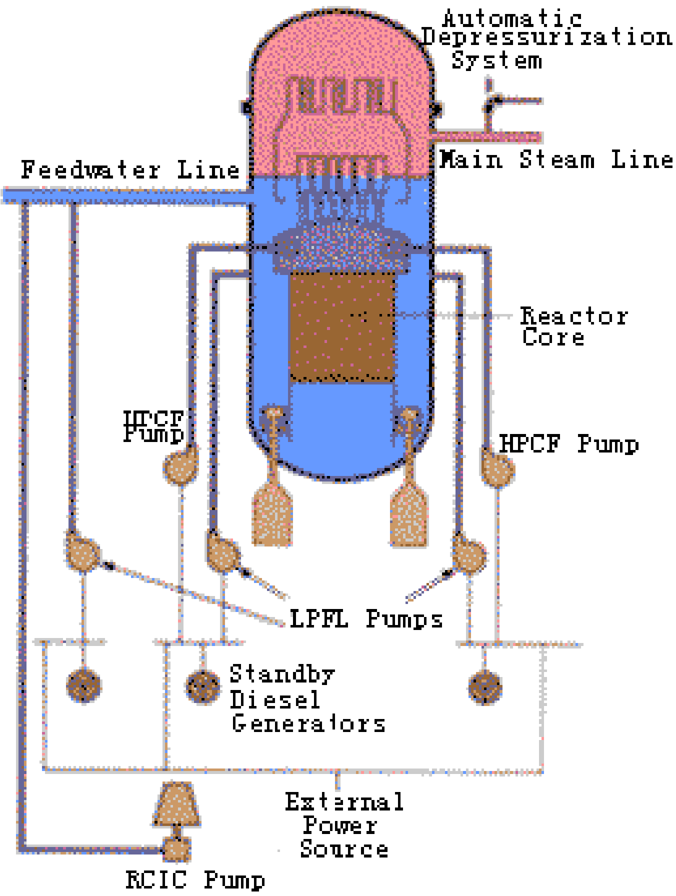


Figure 15. Emergency Core Cooling System (ECCS)

(注)
ECCS: Emergency Core Cooling System
HPCF: High Pressure Core Flooder (System), RCIC: Reactor Core Isolation Cooling (System), LPFL: Low Pressure Flooder (System), ADS: Auto-Depressurization System

The primary containment vessel encloses the reactor pressure vessel, other primary components and piping. In the highly unlikely event of an accident, this shielding prevents the release of radioactive substances. The ABWR uses a reinforced concrete containment vessel (RCCV). Its reinforced concrete outer shell is designed to resist pressure, while the internal steel liner ensures the RCCV is leak-proof. The compact cylindrical RCCV integrated into the reactor building enjoys the advantages of earthquake-resistant design and economic construction cost.

BWR **ABWR**

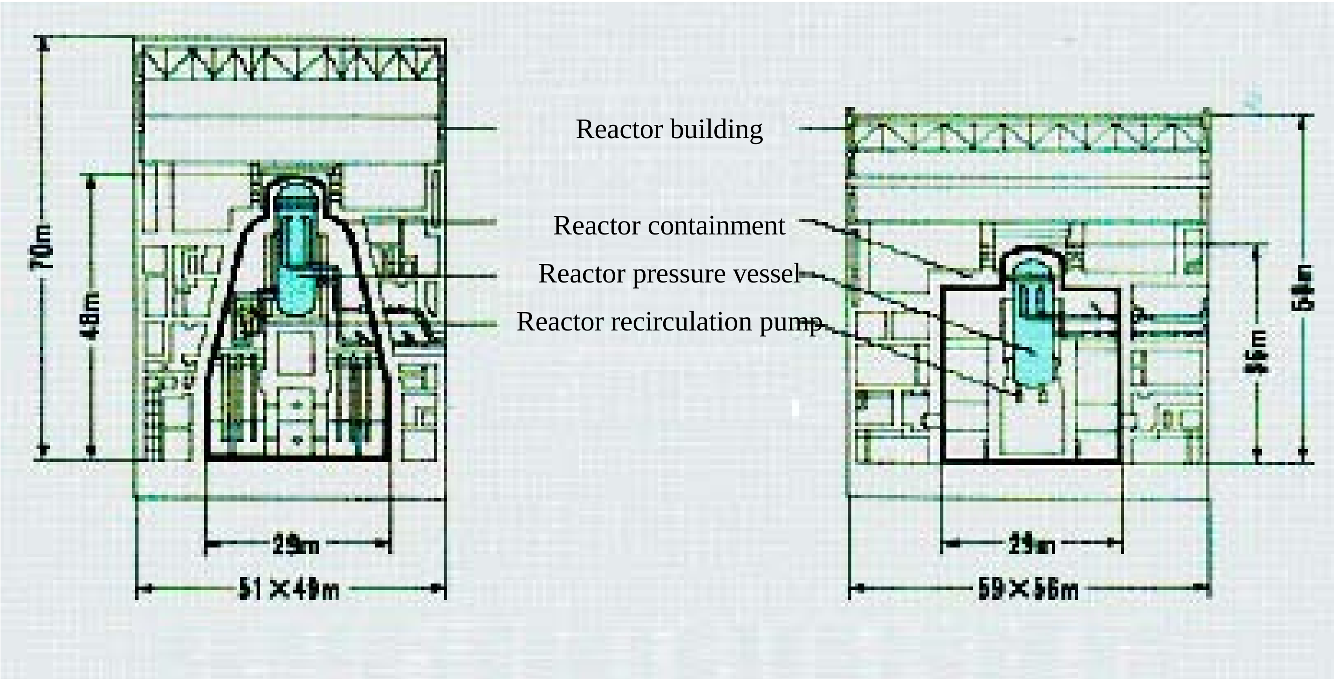


Figure 16. Reactor Containment for BWR and ABWR

(6) Digital control and instrumentation systems

The control and instrumentation (C&I) systems use state of the art digital and fiber optic technologies. The ABWR has four separate divisions of safety system logic and control, including four separate, redundant multiplexing networks to provide absolute assurance of plant safety. Each system includes microprocessors to process incoming sensor information and to generate outgoing control signals, local and remote multiplexing units for data transmission, and a network of fiber optic cables. Multiplexing and fiber optics have reduced the amount of cabling in the plant.

(7) Control room design

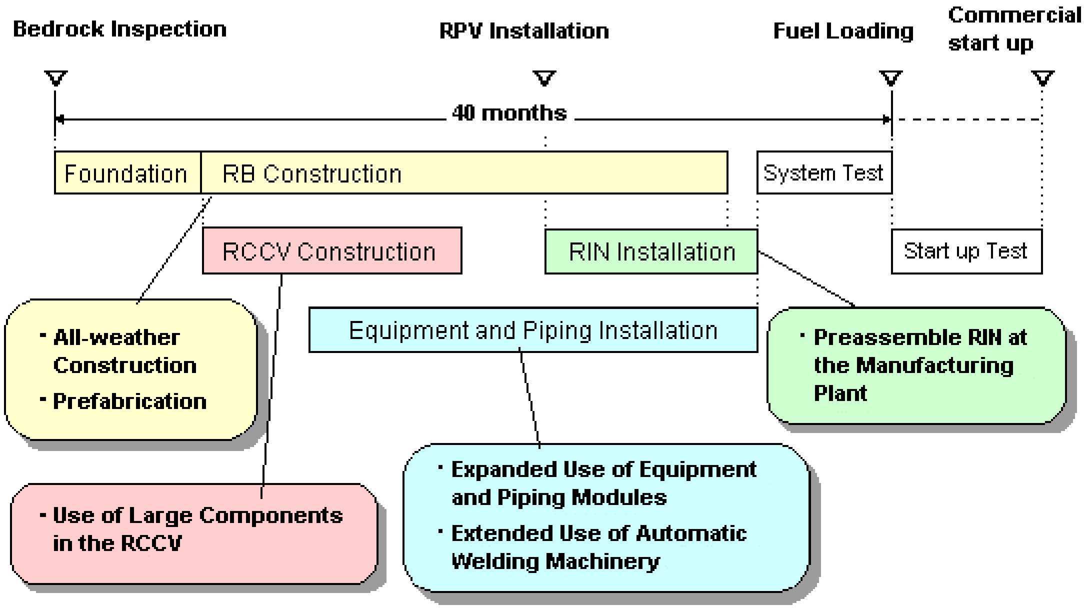
The entire plant can be controlled and supervised from the centered console and the large display panel in the main control room. The left side of console and large display panel is for the safety systems and the right side is for the balance of plant (turbine-generator, feedwater system etc.). The CRTs and flat panel displays on the centered console and the large display panel allow the operator to call up any system, its subsystems and components just by touching the screen. It is possible to operate an entire system in manual operation mode.

Figure 17. Control Room Design

(8) Plant construction

The reactor and turbine building are arranged "in-line" and none of the major facilities are shared with the other units. The containment is a reinforced concrete containment vessel (RCCV) with a leak tight steel lining. The containment is surrounded by the reactor building, which doubles as a secondary containment. A negative pressure is maintained in the reactor building to direct any radioactive release from the containment to a gas treatment system. The reactor building and the containment are integrated to improve the seismic response of the building and the containment are integrated to improve the seismic response of the building without additional increase in the size and load bearing capability of the walls.

At construction of the plant large modules which are prefabricated in the factory are used and assembled to large structure on site. A 1000 ton-crawler crane will lift these modules and place them vertically into the plant. Use of RCCV, modular construction and other construction techniques reduce construction times.



RCCV: Reinforced Concrete Containment Vessel
RPV: Reactor Pressure Vessel
RIN: Reactor Internals
RB: Reactor Building

Figure 18. ABWR construction schedule (typical)

Particular attention was paid to designing the plant for ease of maintenance. Monorails are available to remove equipment to a conveniently located service room via an equipment hatch.

Removal of the reactor internal pumps and FMCRDs for servicing has been automated. Handling devices, which in the case of the FMCRD is operated remotely from outside the containment, engage and remove the equipment. The pump or driver is laid on a transport device and removed through the equipment hatch. Just outside the hatch are dedicated service rooms, one for the RIPs and another for the FMCRDs, where the equipment can be decontaminated and serviced in a shielded environment. The entire operation is done efficiently and with virtually no radiation exposure to the personnel.

Chapter 6. Economic Simplified Boiling Water Reactor (ESBWR)

6.1 ESBWR and Natural Recirculation

The Economic Simplified Boiling Water Reactor (ESBWR) is a passively safe generation III+ reactor which builds on the success of the ABWR. Both are designs by General Electric, and are based on their BWR design. The plant data are shown in Table 4.

Table 4. ESBWR Technology Fact Sheet

Plant Life (years)	60
Thermal Power	4,500 MW
Electrical Power	1,560 MW
Plant Efficiency	34.7 %
Reactor Type	Boiling Water Reactor
Core	
Fuel Type	Enriched UO2
Fuel Enrichment	4.2%
No. of Fuel Bundles	1,132
Coolant	Light water
Moderator	Light water
Operating Cycle Length	12-24 months
Outage Duration	~14 days
Percent fuel replaced at refueling	See footnote 4
Average fuel burnup at discharge	~50,000 MWd/MT
Number of Steam Lines	4
Number of Feedwater Trains	2
Containment Parameters	
Design Temperature	340°F
Design Pressure	45 psig
Reactor Parameters	
Design Temperature	575°F
Operating Temperature	550°F
Design Pressure	1,250 psig
Nominal Operating Pressure	1,040 psia
Feedwater & Turbine Parameters	
Turbine Inlet/Outlet Temperature	543/93°F
Turbine Inlet/Outlet Pressure	985/0.8 psia
Feedwater Temperature	420°F
Feedwater Pressure	1,050 psia
Feedwater Flow	4.55 x 104 gpm
Steam mass flow rate	19.31 x 106 lbs/hr
Yearly Waste Generated	
High Level (spent fuel)	50 metric tons

Intermediate Level (spent resins, filters, etc.) and Low Level (compactable/non-compactable) Waste	1,765 cubic

The ESBWR uses natural circulation with no recirculation pumps or their associated piping.

Through design simplification, natural circulation in GE's ESBWR will decrease Operations and Maintenance (O&M) costs, reducing the overall cost of plant ownership. Natural circulation provides simplification over previous Boiling Water Reactor (BWR) and all Pressurized Water Reactor (PWR) designs that rely on forced circulation. This improvement is accomplished by the removal of recirculation pumps and associated motors, piping, valves, heat exchangers, controls, and electrical support systems that exist with forced circulation. Natural circulation in the ESBWR also eliminates the risk of flow disturbances resulting from recirculation pump anomalies.

The ESBWR and internals is shown in Figure 19. and the natural recirculation of ESBWR is shown in Figure 20.

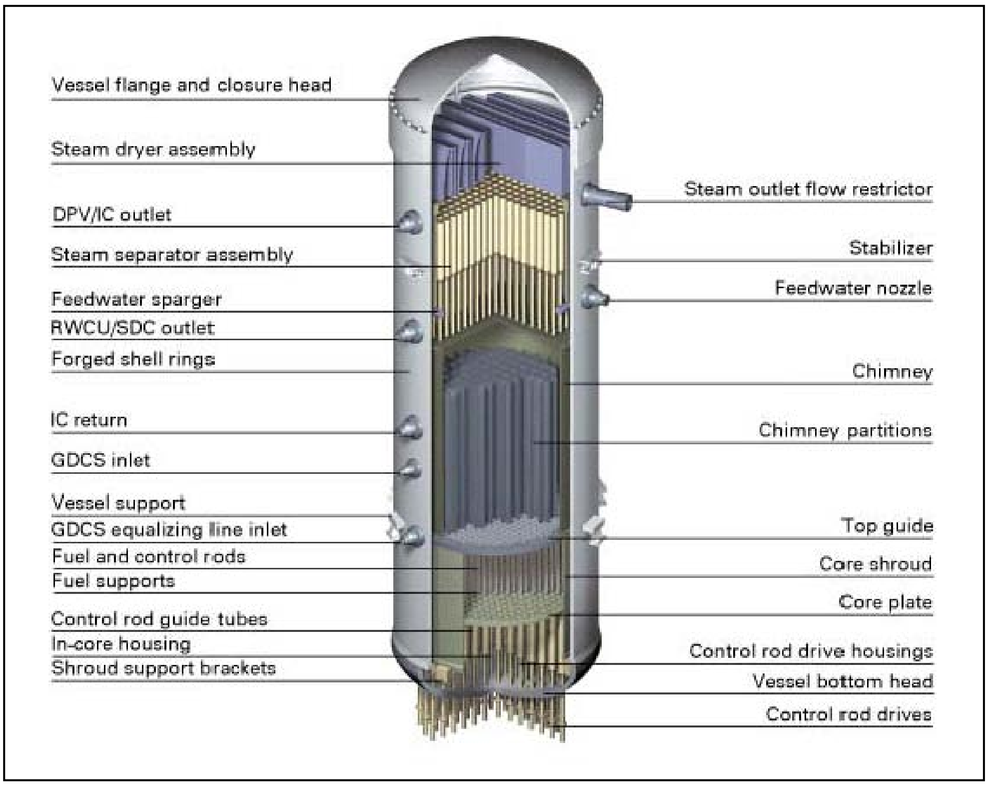


Figure 19. ESBWR and Internals

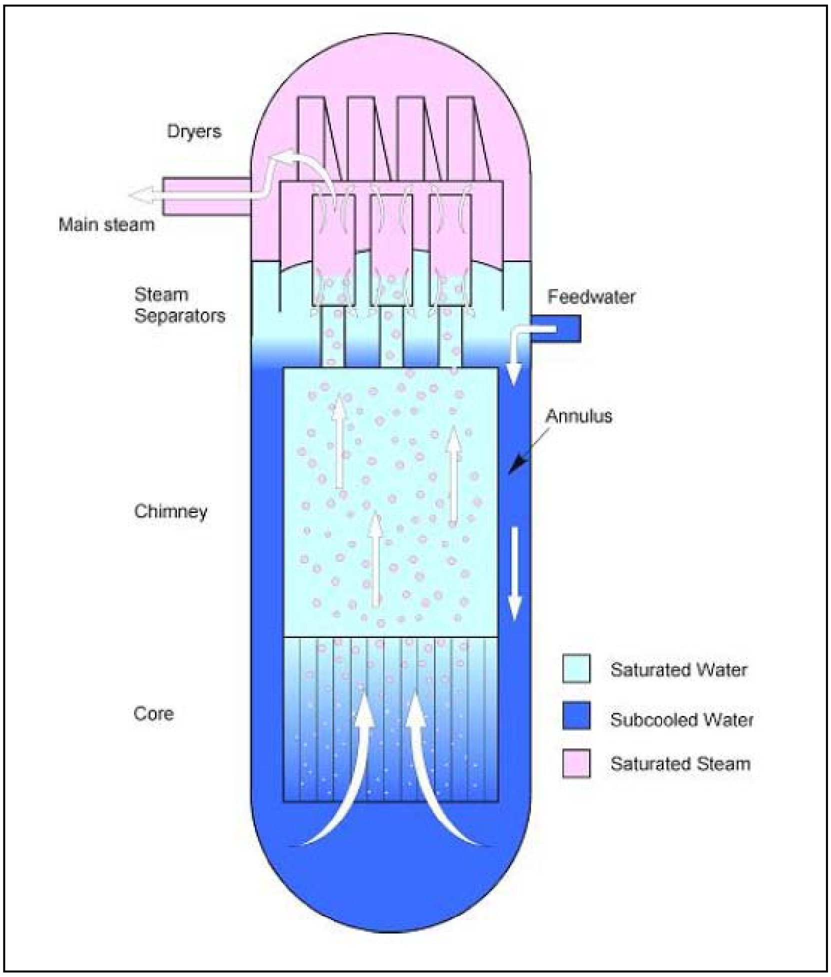


Figure 20. ESBWR Natural Recirculation

Natural circulation is consistent with the key objectives of the ESBWR program: a passive safety design with simplification achieved by evolutionary enhancements. Most of the components in the ESBWR design are standard to BWRs and have been operating in the commercial nuclear energy fleet for years. The main differences between natural and forced circulation are the additions of:

• A partitioned chimney above the reactor core to stabilize and direct the steam and water flow above the core.

• A correspondingly taller, open down-comer annulus that reduces flow resistance and provides additional driving head, pushing the water to the bottom of the core.

Natural circulation is a proven technology. Valuable operating experience was gained from previously employed natural circulation BWR designs. Examples of plants using only natural circulation include the Humboldt Bay plant in California and the Dodewaard plant in the Netherlands, which operated for 13 and 30 years respectively.

Today, large (>1000MW) BWRs can generate about fifty percent of rated power in natural circulation mode. The operating conditions in this mode—power, flow, stability, steam quality, void fraction, void coefficient, power density, and power distribution— are predicted by GE calculation models that were calibrated against operating plant data from LaSalle, Leibstadt, Forsmark, Confrentes, Nine Mile Point 2, and Peach Bottom 2. The ESBWR utilizes proven natural circulation technology to operate a reactor with the size and performance characteristics customers need today at one hundred percent of rated power.

6.2 ESBWR Passive Safety Design

The passively safe characteristics are mainly based on isolation condensers, which are heat exchangers that take steam from the vessel (Isolation Condensers, IC) or the containment (Passive Containment Cooling System, PCCS), condense the steam, transfer the heat to a water pool, and introduce the water into the vessel again.

Those systems are illustrated in Figure 21 and 22.

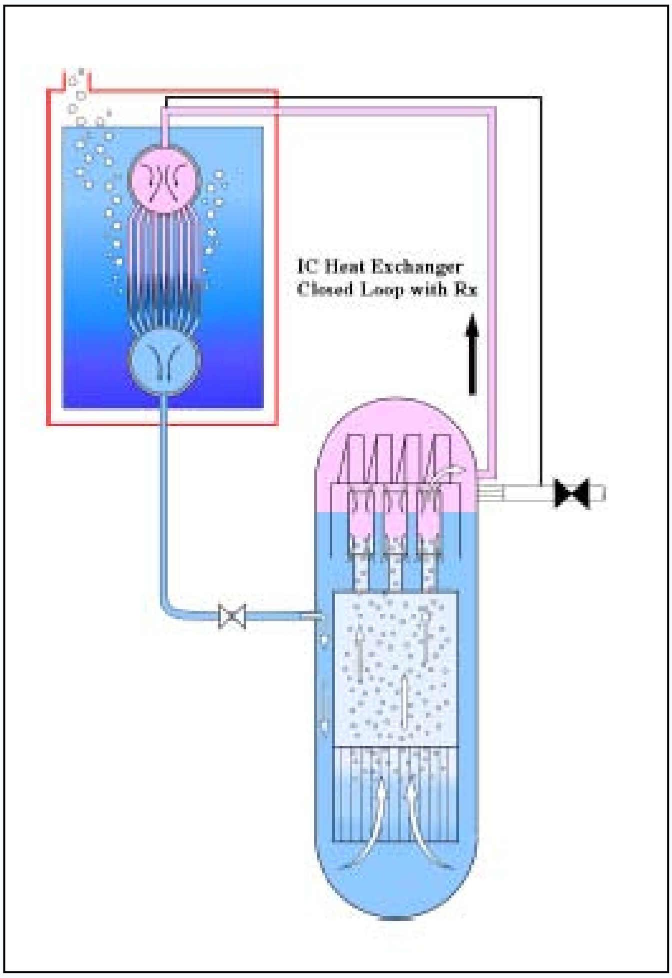


Figure 21. Isolation Condenser System

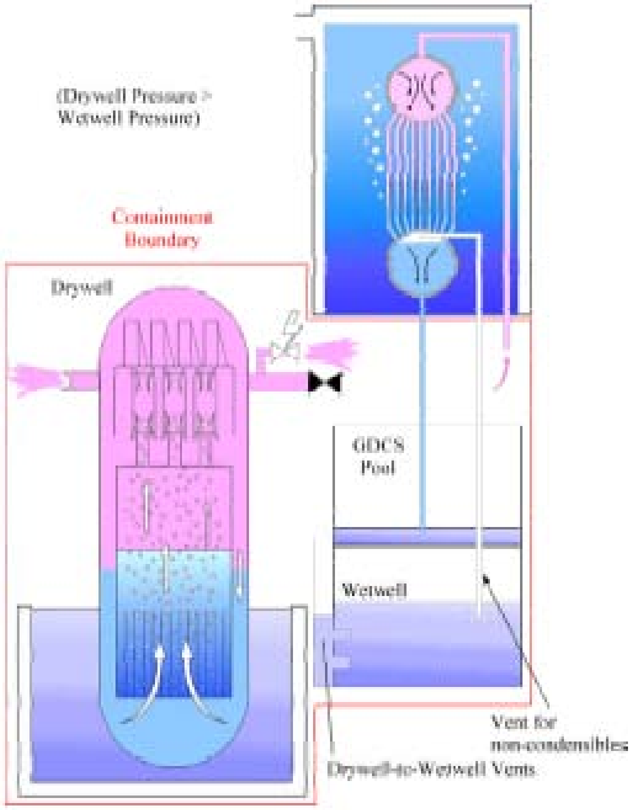


Figure 22. Passive Containment Cooling System

This is also based on the gravity driven cooling system (GDCS) shown in Figure 23, which are pools above the vessel that when very low water level is detected in the reactor, the depressurization system opens several very large valves to reduce vessel pressure and finally to allow these GDCS pools to reflood the vessel.

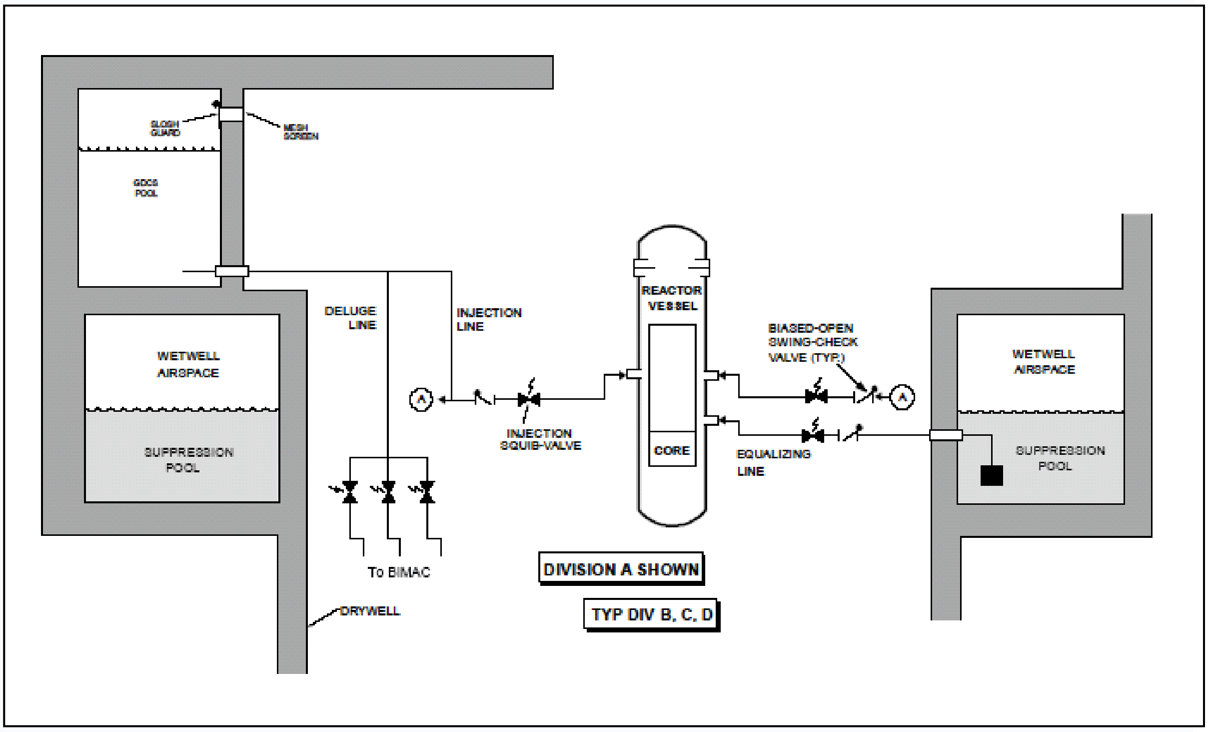


Figure 23. Gravity-Driven Cooling System

The core is shorter than conventional BWR plants because of the smaller core flow (caused by the natural circulation). There are 1132 bundles and the thermal power is 4500 MWth (1550 MWe).

Below the vessel, there is a piping structure which allows for cooling of the core during a very severe accident. These pipes divide the molten core and cool it with water flowing through the piping.

The probability of radioactivity release to the atmosphere is several orders of magnitude lower than conventional nuclear power plants, and the building cost is 60-70% of other light water reactors.

The energy production cost is lower than other plants due to:

1. Lower initial capital cost
2. Lower operational and maintenance cost

General Electric has recalculated maximum core damage frequencies per year per plant for its nuclear power plant designs:

BWR/4 -- 1 x 10^{-5} (a typical plant)
BWR/6 -- 1 x 10^{-6} (a typical plant)
ABWR -- 2 x 10^{-7} (now operating in Japan)
ESBWR -- 3 x 10^{-8} (submitted for Final Design Approval by NRC)

The ESBWR's maximum core damage frequency is significantly lower than that of the AP1000 or the European Pressurized Reactor.

Chapter 7. Current status

As of December 2006, four ABWRs were in operation in Japan: Kashiwazaki-Kariwa units 6 and 7, which opened in 1996 and 1997, Hamaoka unit 5, opened 2004 having started construction in 2000, and Shika 2 commenced commercial operations on March 15, 2006. Another two, identical to the Kashiwazaki-Kariwa reactors, were nearing completion at Lungmen in Taiwan, and one more (Shimane 3) had just commenced construction in Japan, with major siteworks to start in 2008 and completion in 2011. Plans for at least six other ABWRs in Japan have been postponed, cancelled, or converted to other reactor types, but three of these (Higashidori 1 and 2 and Ohma) were still listed as *on order* by the utilities, with completion dates of 2012 or later.

Several ABWRs are proposed for construction in the United States under the Nuclear Power 2010 Program. However these proposals face fierce competition from more recent designs such as the ESBWR (Economic Simplified BWR, a generation III+ reactor also from GE) and the AP1000 (Advanced, Passive, 1000MWe, from Westinghouse). These designs take passive safety features even further than the ABWR does, as do more revolutionary designs such as the pebble bed modular reactor.

On June 19, 2006 NRG Energy filed a Letter Of Intent with the Nuclear Regulatory Commission to build two 1358-MWe ABWRs at the South Texas Project site.

New Reactor Licensing Applications in US including ABWR and ESBWR from 2005 to 2010 and beyond are shown in the Figure 24.

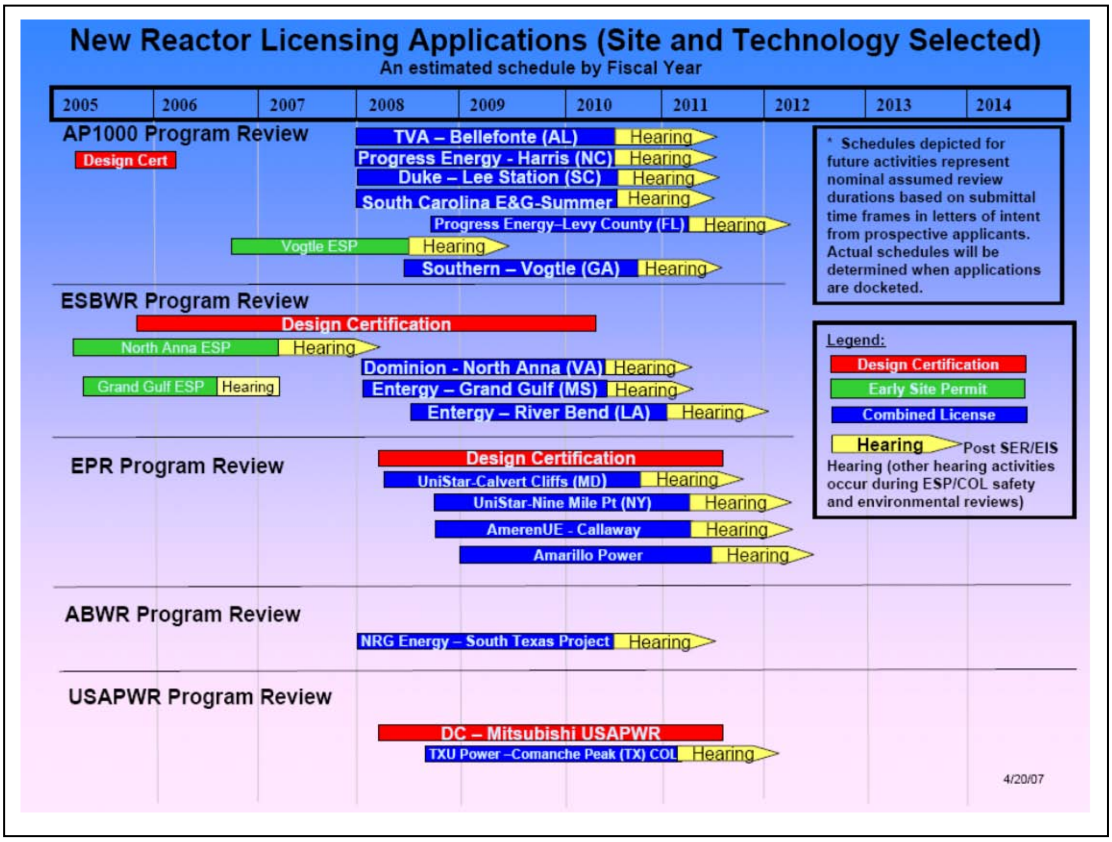


Figure 24. New Reactor Licensing Applications in US

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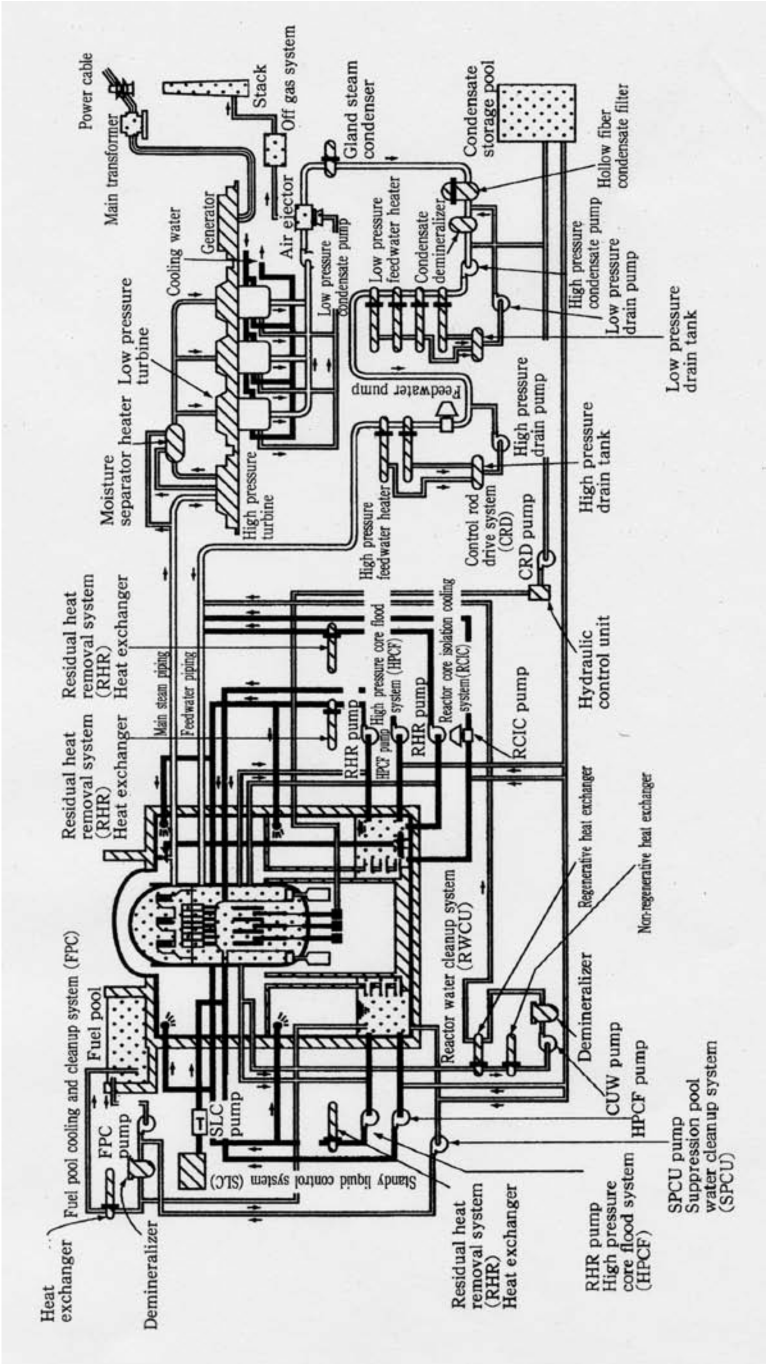
More details on the System Outline of ABWR NPP



Long-term Training Course
on
Safety Regulation and Safety Analysis / Inspection
2005

OUTLINE OF SAFETY DESIGN
(CASE OF BWR)

7 September – 11 November 2005
Tokyo, Japan

Japan Nuclear Energy Safety Organization

(JNES)

Summary of Safety Design of Nuclear Power Station
(Case of BWR)

Contents

1. Concept of Safe Design of Nuclear Power Station

1.1 Three Principles of Safety Ensuring

Safety ensuring of nuclear power station is to prevent irradiation exposure to the public in the neighborhood of the nuclear power station.

The fundamental rules of safety ensuring are the following three principles, which are commonly called the three principles of safety ensuring.

- Reactor shutdown
- Reactor cooling
- Radiation confinement

1.1.1 Reactor shutdown

When any anomaly occurs during reactor operation, the reactor is shutdown to prevent further nuclear fission.

BWR is provided with two different systems to shutdown the reactor; one is the reactor shutdown system, which stops nuclear fission by inserting control rods into the reactor core. The control rods contain neutron absorbing materials. The other is the boron insertion system, which stops nuclear fission by injecting boric acid solution into the reactor core.

1.1.2 Reactor cooling

Even after the nuclear fission is stopped, the reactor releases decay heat from fission products in the core. Reactor cooling after shutdown is necessary, because the fuel assembly will be damaged by increasing in-core temperature, if the decay heat is not removed.

BWR is provided with a residual heat removal system, which consists of motor driven pumps and heat exchangers.

1.1.3 Radiation confinement

If radioactive materials leak from the core, the radioactivity must be confined to prevent outside leakage from the nuclear power station.

The radiation confinement is the ultimate means to protect the public in the neighborhood against radiation exposure. For this effect, the following 5 barriers are provided.

- Barrier 1. Fuel pellet matrix

The matrix itself of the fuel pellet, which is ceramic made of sintered UO2 powder, has an excellent ability to retain fission products (FP) and, thus, the totality of the solid FP and most FP gases are contained in the matrix. The fuel temperature during operation is properly controlled to prevent FP from being melted and discharged outside.

- Barrier 2. Fuel cladding

Fuel cladding confines gaseous FP and is made to ensure its integrity by preventing damage due to the lack of coolant or interaction between fuel pellet and cladding.

At the occurrence of an accident involving a loss of coolant, ECCS will be activated to take over the cooling process and to prevent fuel from being damaged or melted.

- Barrier 3. Primary coolant pressure boundary

When a fuel cladding is accidentally broken and FP is discharged into the coolant, FP will have to be contained in the reactor coolant. The primary coolant pressure boundary should be reliable for that purpose.

- Barrier 4. Primary Containment Vessel

When FP is discharged as the pressure boundary is destroyed in an accident involving the rupture of piping, the containment will act as a barrier against the dispersion of radioactivity. With its excellent hermetic ability, the primary containment can significantly reduce the quantity of FP to be discharged.

- Barrier 5. Secondary containment facilities, etc.

Leaking FP from the primary containment vessel-will be retained within the reactor building and infiltrated so that radioactivity can be further diluted before reaching the neighborhood population.

1.2 Defense in Depth

Although there are many barriers to obstruct the dispersion of radioactive materials before they reach the surrounding environment, it is essential to ensure that these barriers effectively function without being destructed. Moreover, they should be well established in ultimate depth of their bases. A design concept called "Defense in Depth" is applied to achieve the objective, which represents a number of defense systems established in stages at the respective levels.

(1) First Level (Prevention of abnormal condition occurrences)

The primary requirement to realize the purpose of safety is to prevent the occurrence of abnormal conditions before they occur actually. The following measures are to be taken to prevent them:

(i) Each device and system shall be designed to demonstrate specified performance with ample margin, and carefully inspected and maintained to prevent any failure or damage under elaborate quality control and quality assurance systems.

(ii) The nuclear reactor core is to have negative reactivity feed back as an inherent safety feature to the operational range. It is therefore impossible for a reactor to run out of control even in case of a sudden reactivity addition, since an inherent negative reactivity will be automatically applied to suppress the increase of reaction.

(iii) The reactor shall be designed not only to maintain the integrity of the primary coolant boundary but also to confine the nuclear and thermal parameters of the core within the specified tolerable limits to prevent fuel damage despite abnormal transient changes that may occur when the reactor is operating. Therefore, no radioactivity would be discharged out of a reactor even if a failure had occurred in any of the main components or system since none of the fuel would be damaged because of the failure. Furthermore, the abnormal transient change would soon dissipate and the disorder would not be sustained for long.

(iv) The cooling boundary of the reactor is designed to have enough strength to endure postulated load, while the materials is selected so that a failure such as stress corrosion cracking can be effectively prevented.

(v) It is designed to prevent a small disorder such as a minor leak of coolant from the reactor cooling pressure boundary by detecting it in advance by a leak detection system to prevent it developing into a serious incident.

(2) Second Level (Prevention of accident Propagation)

Measures shall be taken to detect any disorder such as any kind of a failure or operational failure that may occur during reactor operation, to restore it at an early stage or to prevent it from progressing to any serious degree.

(i) Safety and protection system shall be provided in order to promptly detect occurrence of an abnormal event and automatically activate a reactor shut-down system and engineered safety failures.

The occurrence of an abnormal event can be detected by measuring various parameters that may change pursuant to the sudden rise of neutron flux levels in the reactor, a drop of the water level or pressure of the reactor or increase in the containment pressure.

(ii) The safety protection system shall be designed to have redundancy and independence, so that it will not lose function even at the occurrence of a single failure or loss of any the components or channels that comprise the system. Moreover, it should be able to eventually resume the stable condition and to perform the specified function under disadvantageous conditions such as the loss of its power source or the shutdown of the system.

(iii) The Safety protection system is designed, in principle, so that it can be tested even during operation.

As described above, the occurrence of an abnormal condition will be accurately detected and the propagation of the event will be effectively prevented.

(3) Third Level (Mitigation of the Impacts of Abnormal Event)

Various engineered safety facilities are provided in order to ensure the safety of the population in surrounding areas by preventing the propagation of the accident and reducing its impact, even in an accident which may break out despite all the above precautions. The engineered safety facilities of BWR consist of emergency reactor cooling, primary containment vessel and other related facilities. In a loss of coolant accident, which is the severest postulated accident for a nuclear plant, the engineered safety facilities will be activated to effectively mitigate the impact of the accident.

(i) Assuming a possibility that the reactor core may be exposed to air after the loss of coolant, ECCS-Emergency Core Cooling System-is designed to be initiated under such emergency situation to protect fuel rods from damages by cooling the core for an extended period.

(ii) The decay heat to be generated in the core for a long period of time shall be effectively removed out of the system with RHR-Residual Heat Removal.

(iii) Emergency power supplies shall be installed in the premises of the power station to secure the functions of the engineered safety facilities, if the outside supply is lost.

(iv) Containment shall be constructed to contain all components and systems of the reactor cooling boundary in order to confine radioactivity that may leak from the reactor and to decay over an extended period.

(v) Safety facilities such as containment spray system shall be installed to decrease the pressure by condensing steam inside the reactor containment and actively remove radioactive iodine suspending in the inside-PCV atmosphere by utilizing the scrubbing effect of the spray-system.

(vi) Flammability Gas Control System shall be installed to be operated as necessary on the assumption that flammable gas tends to accumulate in the Primary Containment Vessel (PCV).

(vii) Radioactive gas leaked from PCV shall be released into air from the main stack of ventilation system installed after radioactive iodine is removed by the filter.

1.3 Single Failure Criteria

The single failure criteria shall be applied to the design of systems related to the safety of the nuclear power plant.

The criteria demand a design with due considerations to the redundancy or diversity of the systems, to secure their required functions in the case of single component failure.

In case of active components such as pumps or motors, the criteria are applied to the events of both shorter and longer durations, while the events of longer duration are subject to the criteria in case of static components such as piping.

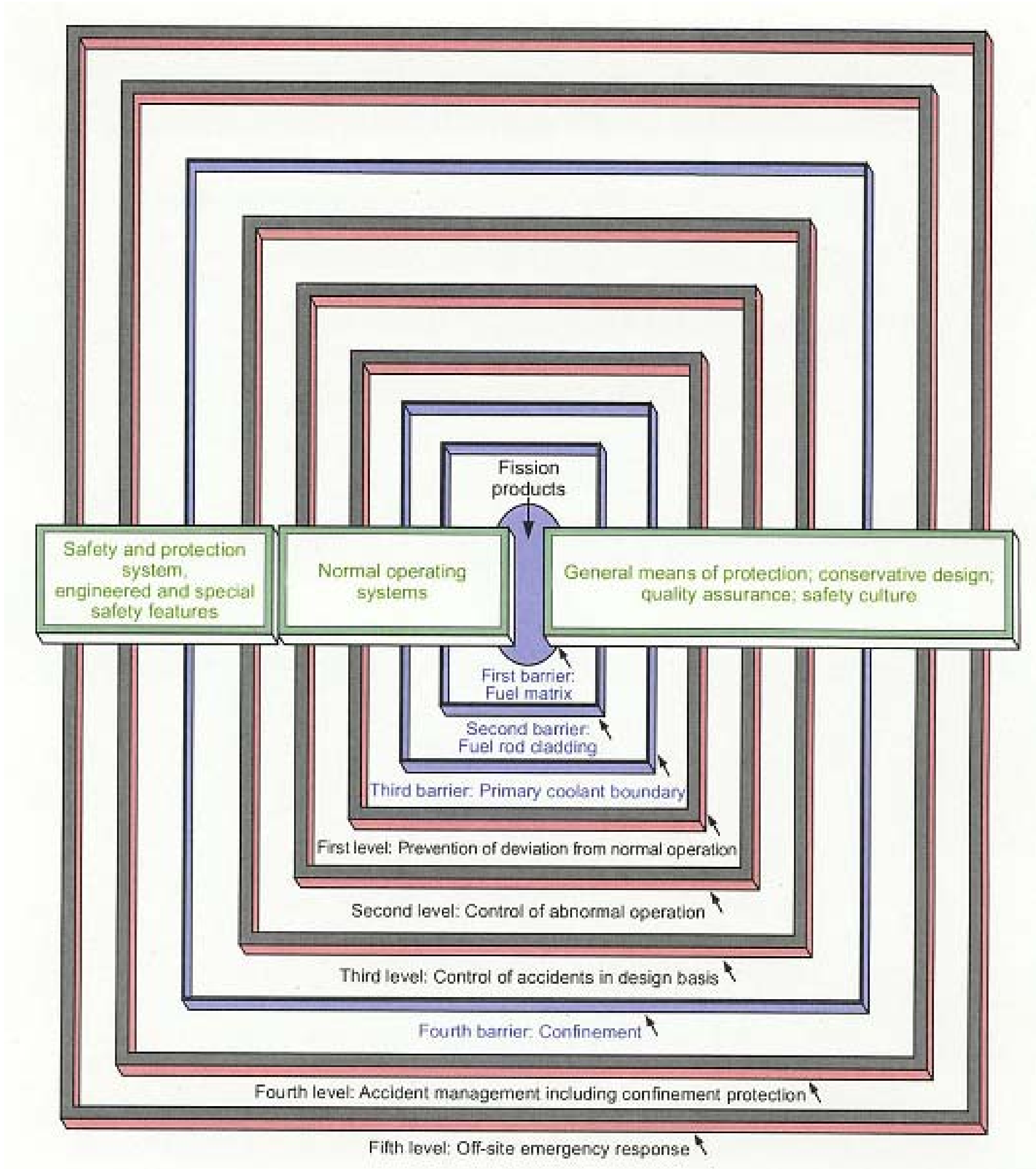


FIG.1.1 The relation between physical barriers and levels of protection in defence in depth
(From IAEA　INSAG-12)

2. Considerations related to the safety design of Reactor and Reaction Control Facility

2.1 Reactor

(1) Fuel

A fuel rod used for the light water reactor has a cladding made of zirconium alloy and is packed with sintered pellets of low enriched uranium dioxide. A part of the gaseous fission products generated by nuclear fission will be released from the pellet while most of the solid elements will remain in the pellet so that the pellet itself can be considered the first barrier to contain the radioactivity. Gaseous fission products released from the pellet will be sealed in the cladding. The fuel cladding is the second barrier to prevent the spread of fission products and is designed and manufactured with sufficient consideration to its quality integrity.

Fuel rod needs to be designed not to exceed the design limit of fuel during normal operation as well at the time of abnormal transient changes. It should be designed so that the maximum linear heat generation ratio will be maintained at 44kw/m or below in order to prevent the center of fuel pellet form reaching the melting point and the core will be operated with the critical power ratio of 1.20 or lower

(1) Reactor core

The reactor core shall be maintained critical where the generation and dissipation of neutrons is held in equilibrium, with reaction of nuclear fission corresponding to the specified output. It is controlled with the control rod possessing negative reactivity. The reaction, however, also depends on such parameters as the fuel temperature, coolant temperature and the volume of void.

The degree of changes in the level of reaction that correspond to the unit quantity of the parameter change is called the reactivity coefficient. The power output coefficient, which combines the temperature reactivity coefficient, moderator temperature coefficient and void coefficient, is designed to be negative in view of safety. This way, the self-controllability is ensured and is useful for preventing the occurrence of the abnormal conditions in the first level of "Defense in Depth" protection.

The ability to maintain the reactor in the subcriticality during operation is the ability to shut down the reactor. The reactor is designed to be able to achieve the stage of subcriticality with a sufficient margin even if one control rod of the highest reactivity had been stuck and could not be re-inserted.

The reactor will be scrammed (emergency shut-down) by the automatic activation of the reactor shut-down system if there is any risk of the reactor reaching the design limit of fuel tolerance

2.2 Reactivity Control Facility

The Reactivity Control Facility of the Reactor has a function of controlling the reactor output in order to prevent the abnormal condition, which is the first level of the Defense in Depth, and a function to shut down the reactor in order to prevent the propagation of the abnormal condition, which is the second level. It consists of the control rod, control rod drive system and the standby liquid control system, which injects bordic acid solution within 30 minutes when the control rod is not inserted.

The neutron flux in the reactor core is controlled by driving up and down the control rod containing neutron absorbing material. The control rod will be immediately inserted within a few seconds into the reactor core to shut it down at the time of emergency.

The considerations for safety are as follows:

(i) Even if a control rod was removed down out of the reactor core for some reason, dropping speed will be controlled so that a sudden application of the reactivity will be restricted.

(ii) The control rod can be quickly inserted in order to prevent a damage to occur to a fuel rod at the time of abnormal transient or accident and also to shut down the reactor by scramming at the time of an earthquake.

(iii) Measures are taken to prevent the control rod from falling down due to a rupture being caused to the control rod drive or its housing.

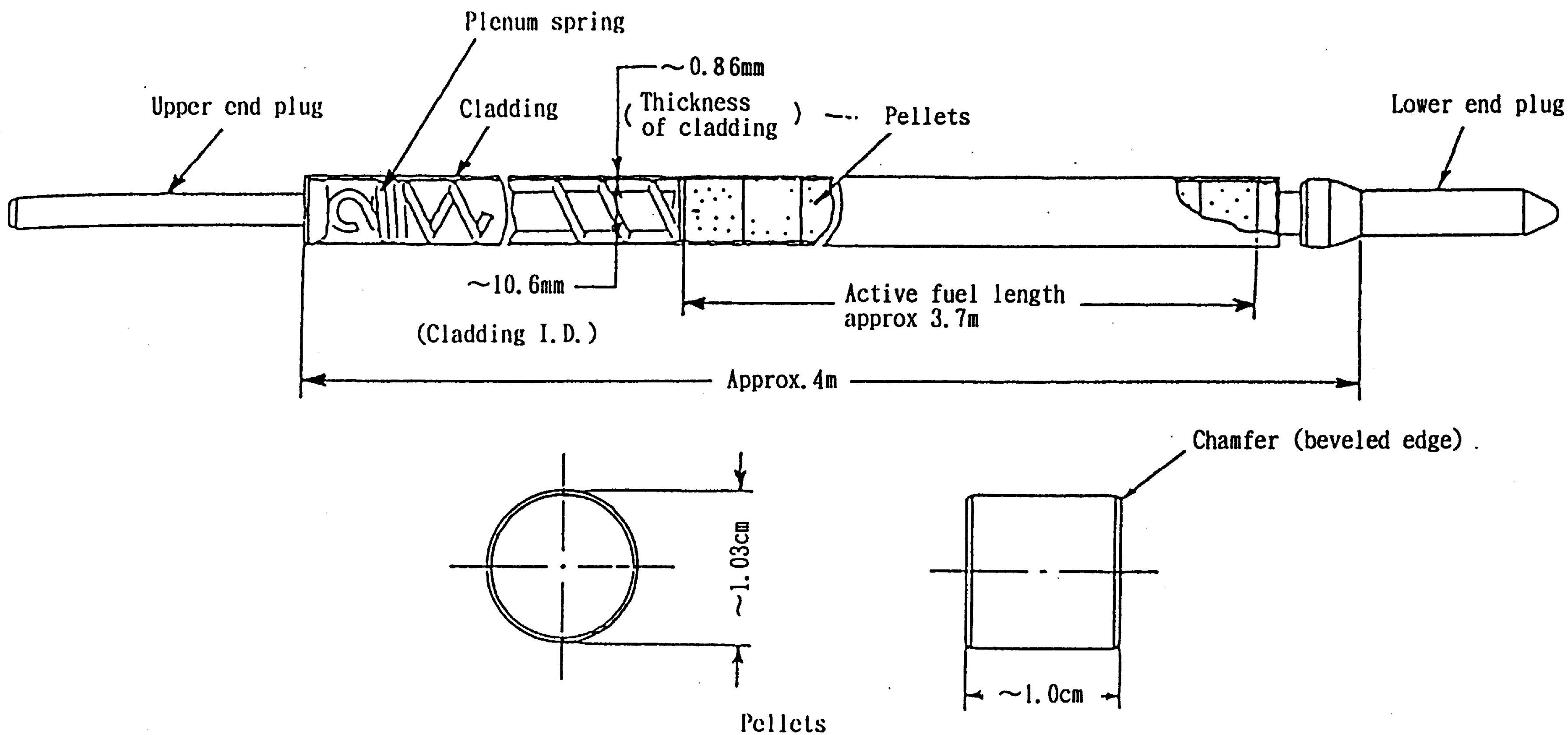


Fig.2.1 Fuel Rod

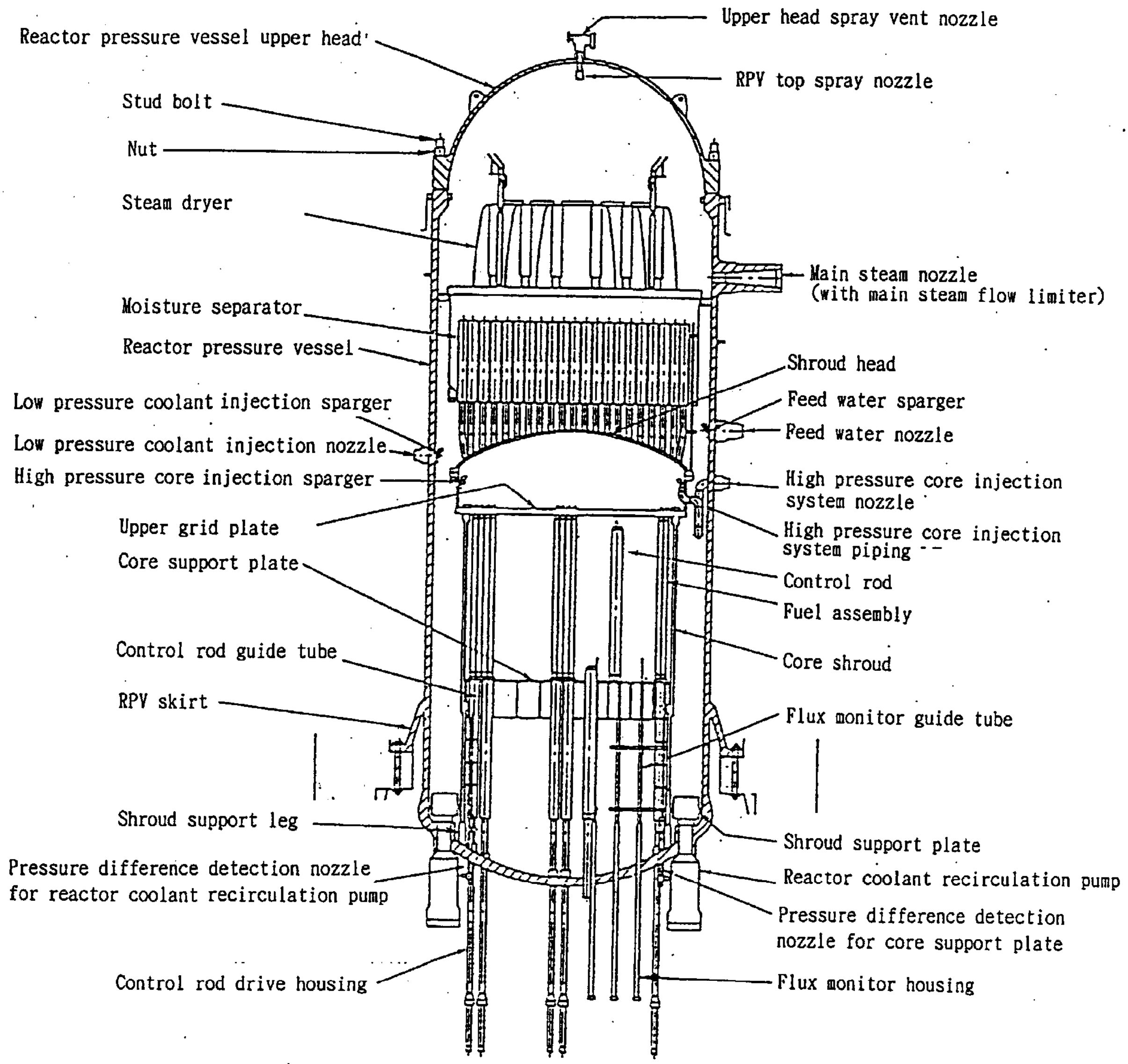
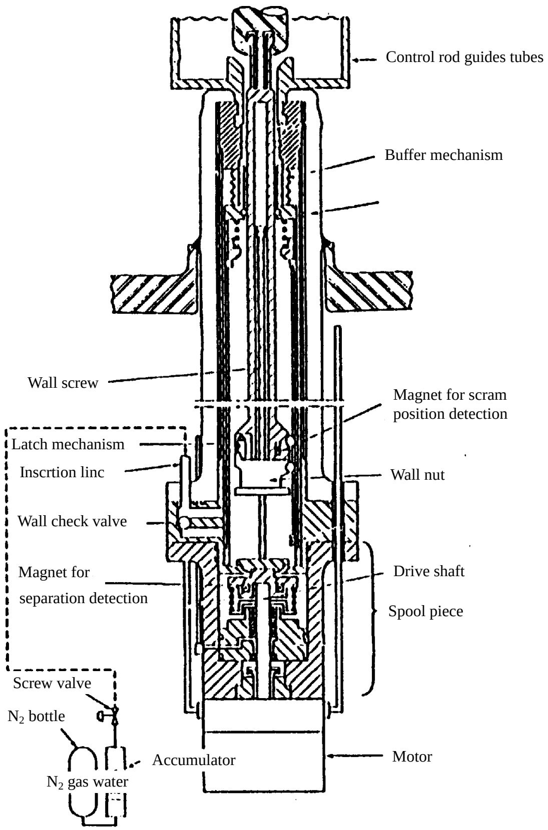


Fig.2.2 Advanced Boiling (Light) Water Reactor (ABWR)



Schematic structure of control rod drive mechanism

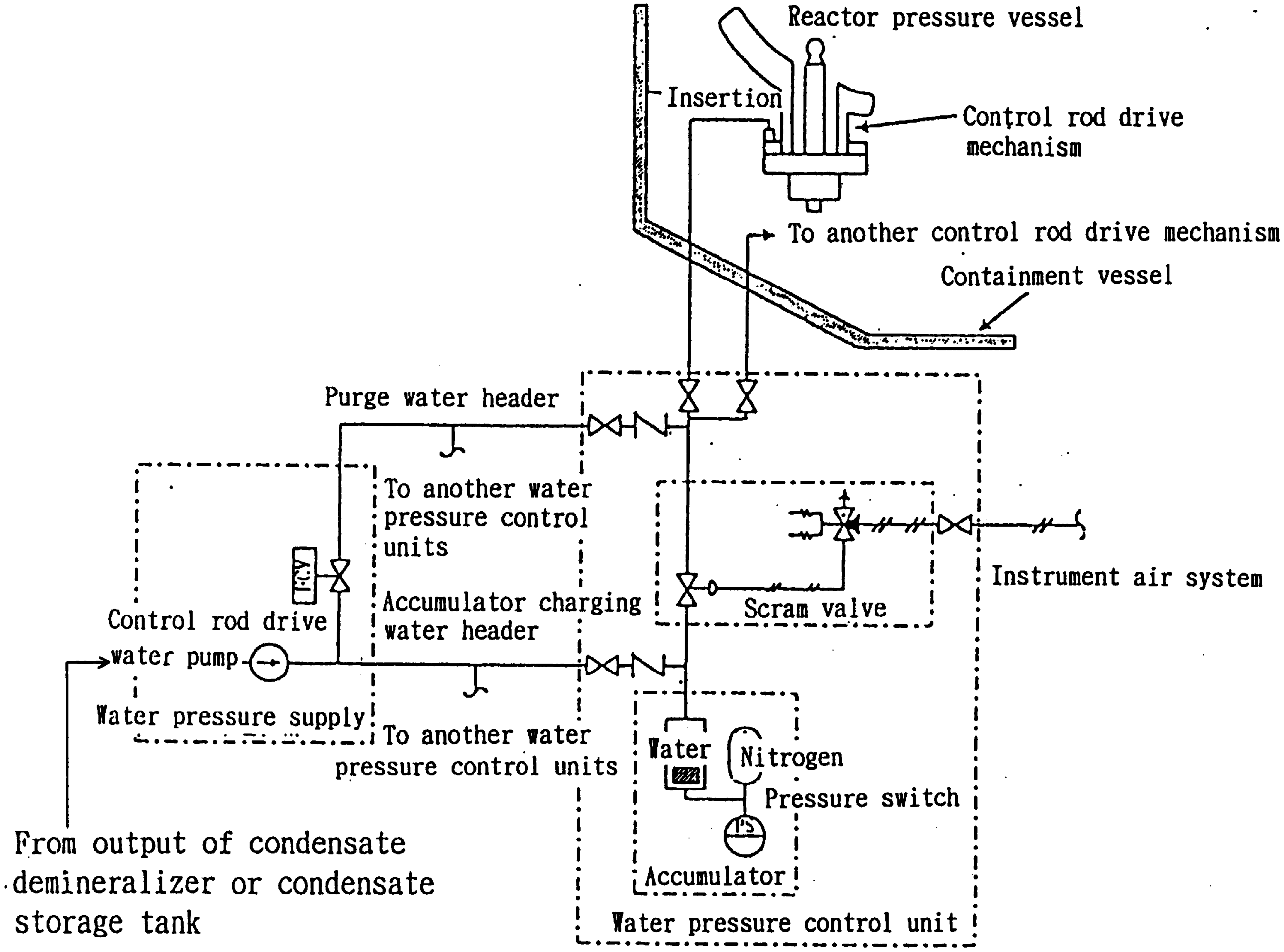


Fig.2.3 Control Drive System

3. Integrity of the Reactor Coolant Boundary

The reactor coolant pressure boundary is the third barrier to prevent the release of fission products (FP) provided next to the second barrier of fuel cladding. It is one of the most important factors to ensure the safety of the reactor facility because the coolant would be released if the boundary was ruptured.

The reactor coolant boundary comprises the reactor pressure vessel, a portion of the feed water system piping, a portion of main steam system piping and portions of piping and valves of other auxiliary system. Considerations are made to minimize the possibility of abnormal leakage of coolant or a loss or damage of the system by selection of proper materials, aseismatic design, and the prevention of excessive pressure, etc. Furthermore, arrangements are made to allow inspection of the reactor coolant boundary to be performed during the maintenance period while the reactor is shut down for refueling or any other reason. It is also analytically verified that the system is designed to have enough strength even in design base transient events.

For the components made of ferrite steel, due considerations are made when operating to ensure that no sudden propagating ruptures (unstable breakage) owing to brittleness would occur in the reactor coolant boundary under any operating conditions. For example, the operation of the reactor, while the temperature of the primary system is increasing or decreasing, is performed by limiting heating rate or cooling rate, and the reactor is operated by keeping itself in the safety operational range.

In a reactor pressure vessel, in particular, considerations are done in terms of material selection, design and fabrication process. The possibility of the transition temperature increase in brittleness is also taken into consideration when radiated by fast neutron. Furthermore, the minimum operating temperature, operating restrictions and other conditions are to be determined by testing with the use of the test specimen, which are fabricated from the same reactor pressure material and placed around the reactor core in capsule for a prescribed time.

It is generally recognized that a leakage can be normally detected before a crack propagates to a breakage (LBB). Monitoring and detecting systems of different measurement principles such as radioactivity monitor, the sump water system, condensation liquid measuring system are installed to monitor the leakage of the primary coolant from the reactor coolant boundary.

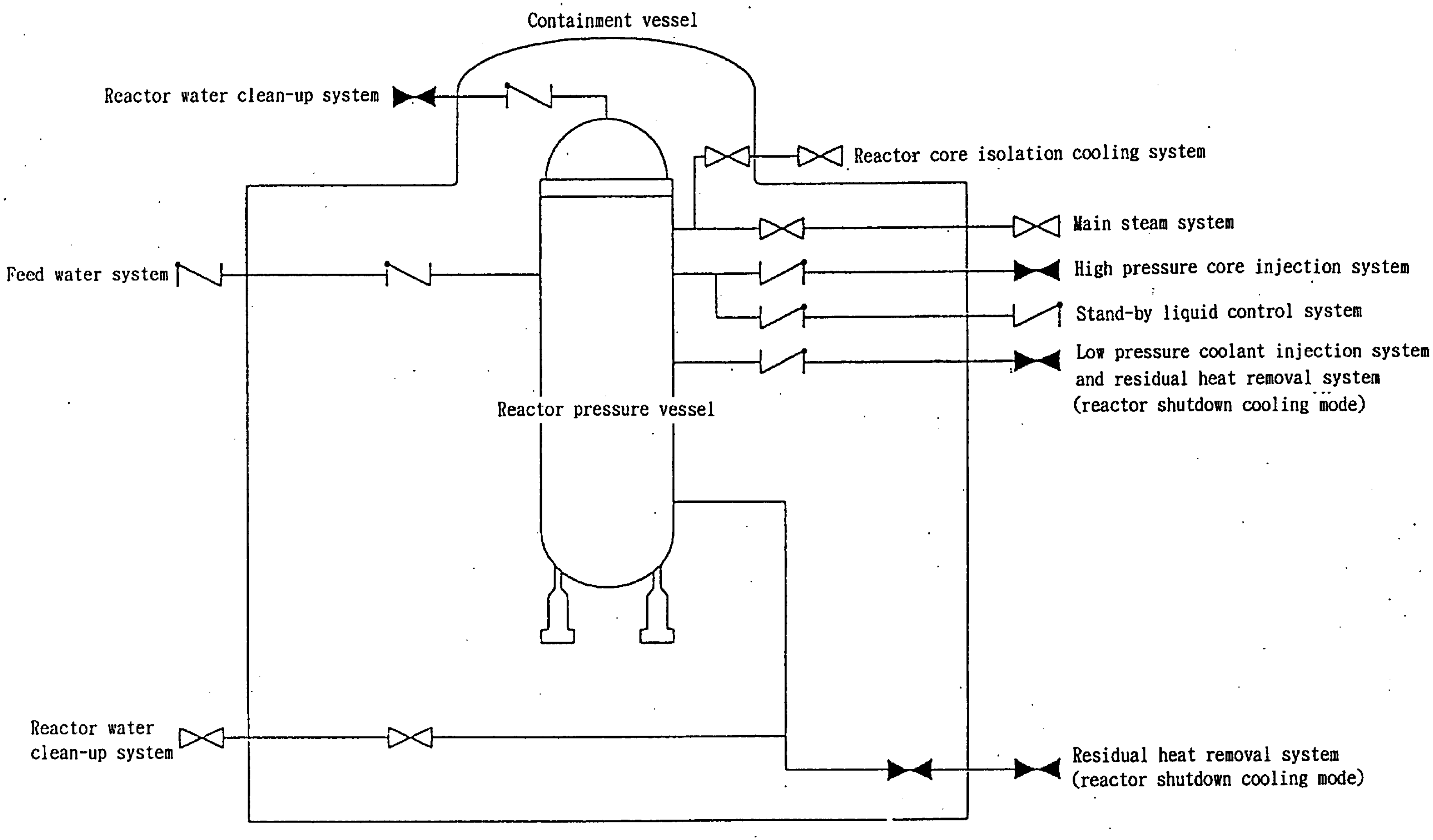


Fig.3.1 Reactor Coolant Pressure Boundary

4. Emergency core cooling system

The emergency core cooling system is provided to mitigate abnormal events, based on the concept 'Defense in Depth Level No. 3', and designed so as to satisfy the following requirements:

4.1 Design Criteria

(1) Single failure criteria and loss of external power source

In the design of ECCS, single failure criteria (Design concept 'Redundancy' should be adopted so that the system functions as specified, even if a single component fails.)is applied for active component in short and long term and for passive component in long term. Furthermore, measures should be taken against the case then the external power source is lost.

(2) Physical separation of trains

In satisfying the above single failure criteria, a component or a system is to be divided into plural small components or subsystems(trains) with the similar configuration, thus realizing redundancy. In ABWR, a design concept of 3 subsystems(trains) is adopted and the subsystems are arranged to be physically separated one another to protect the impact of damages in the other subsystem (3 division).

(3) Cooling Criteria

To prevent expended failure of fuel clad and maintain the core configuration even in case of LOCA, both fuel clad temperature and localized water-metal reactivity must be limited to the following degrees:

Max. temperature of fuel clad 1,200℃

Localized water-metal reactivity Less than 15 percent of sheath tube thickness

(4) Criteria for removal of core decay heat over a long period

The capacity for core cooling on a long term must be provided with the core configuration being maintained as it is, to remove decay heat of long half-life radioactive nuclide.

(5) There are two alternative cooling system: cooling by spray of water into the core and cooling submerging the core in water to keep down any subsequent temperature rise. The ABWR has selected the latter; the submerged cooling system.

4.2 System configuration

(1) Low pressure flooder (LPFL)

The LPFL is put into service when the pressure in the reactor pressure vessel is depressurized rapidly due to the accident of large diameter of rupture. It is designed to inject a large quantity of water to the outside of the reactor core shroud, by use of the low pressure injection mode of the residual heat removal system (RHR), taking water from the suppression pool.

In addition, the LPFL serves to maintain the submerged condition by water injection into the core, in combination with the other systems such as the high pressure core flooder (HPCF), reactor core isolation cooling (RCIC) and automatic depressurizing system (ADS).

The system is composed of three (3) motor-driven pumps, piping and valves. They are automatically actuated at the signals indicating 'reactor water level low (Level 1)' or 'dry well pressure high'. After recovery of the water level, large amount of water is not necessary and only making up for evaporation is required.

(2) High pressure core flooder (HOCF)

The HPCF, consisting of the motor-driven pumps, piping and valves. works functionally to cool down the core in combination with the LPFL, RCIC and ADS. The system starts its actuation at the signal of 'reactor water level low (LEVEL 1.5)' or 'dry well pressure high'. It is capable of core cooling by injection of water over the fuel assembly through the Spurger nozzle on top of the core, even if the reactor pressure remains relatively high. It has the primary and secondary water sources available for that purpose: as a primary source at first, water is taken from the condensate storage tank of primary source and if it is used up, water in the suppression chamber pool as a secondary source to perform is taken. Water feed can automatically stop at 'reactor water level high (Level 8)' after recovery of the water level.

(3) Reactor core isolation cooling system (RCIC)

In case of the ABWR, the RCIC system is designed with the function of ECCS to strengthen the high pressure system functionally. It is capable of core cooling, same as in the case of the HPCF, should any loss-of-coolant accident take place due to rupturing of the small-size piping. As one of the high pressure cooling system, it constitutes three independent systems in combination with two (2) HPCF trains.

(4) Automatic depressurizing system (ADS)

The ADS is functionally capable of core cooling, coupled with the LPFL system, when the HPCF system fails to work in case of any accident from medium or small rupturing. Upon receipt of both signals indicating 'reactor water level low (Level 1)' and 'dry well pressure high', the system can actuate, with a time lag of 30 seconds, eight (8) out of total 18 relief valves on main steam pipeline. By relief of reactor steam into the suppression chamber and rapid decrease of its pressure, it becomes possible to inject water into the core through the LPFL.

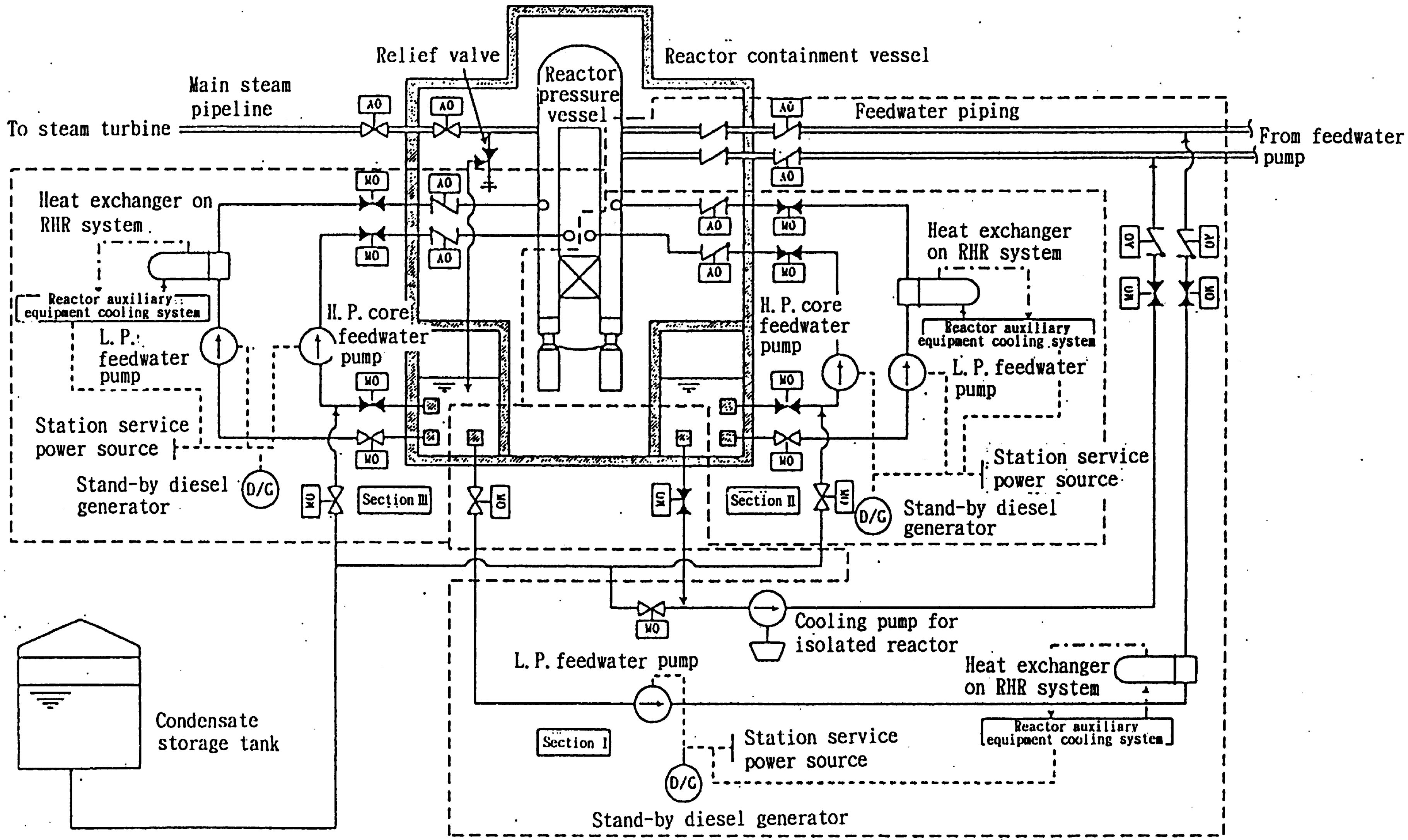


Fig.4.1 Schematic Diagram of Emergency Core Cooling System

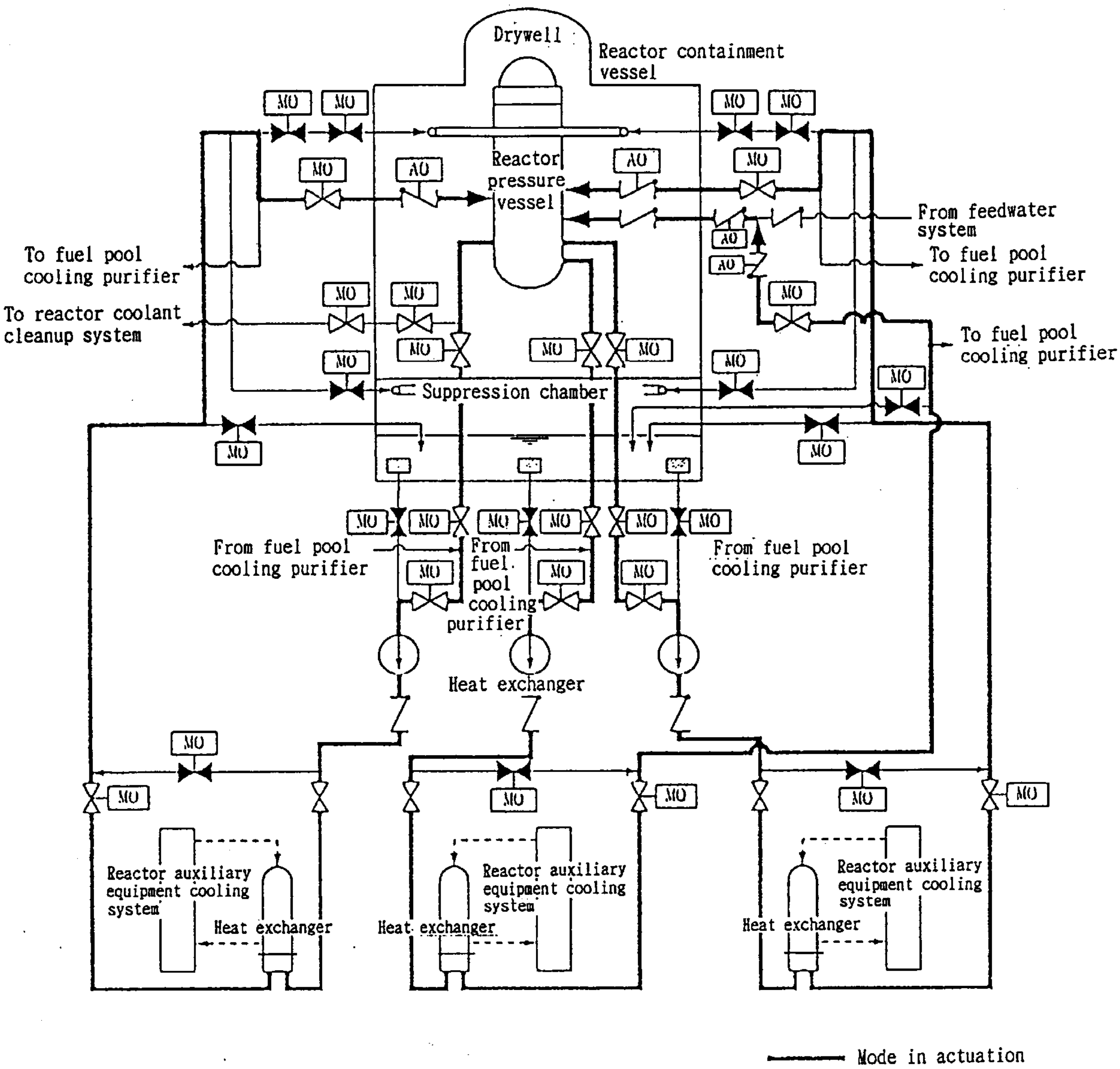


Fig.4.2 Low Pressure Coolant Injection System (LPCI)
(Schematic flow diagram of residual heat removal system at low pressure coolant injection mode operation)

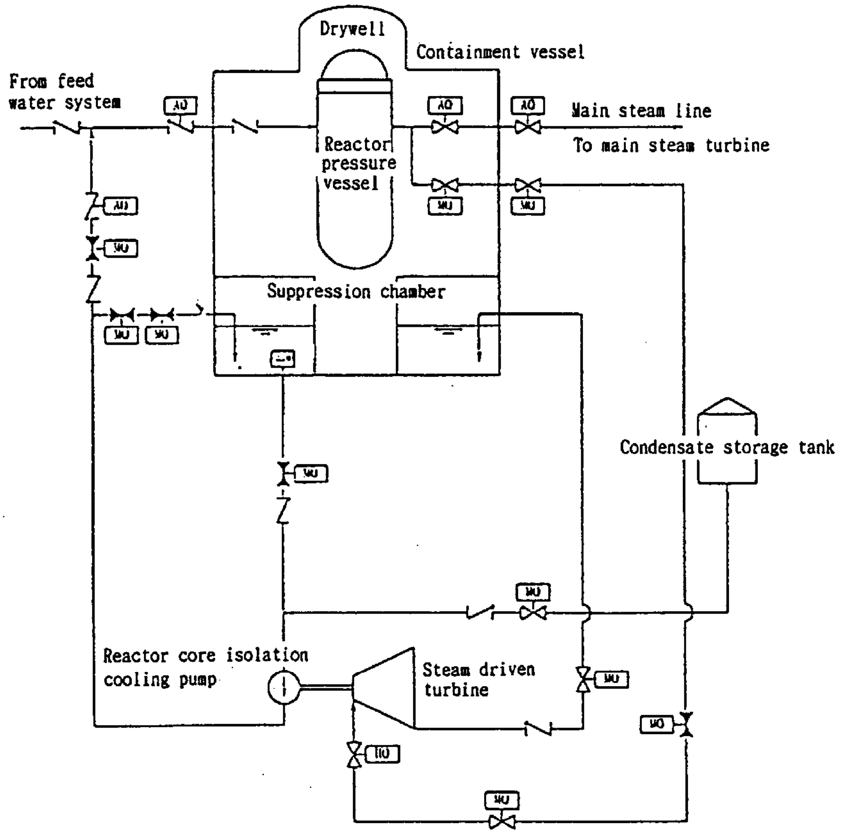


Fig.4.4 Schematic Flow Diagram of Reactor Core Isolation Cooling System

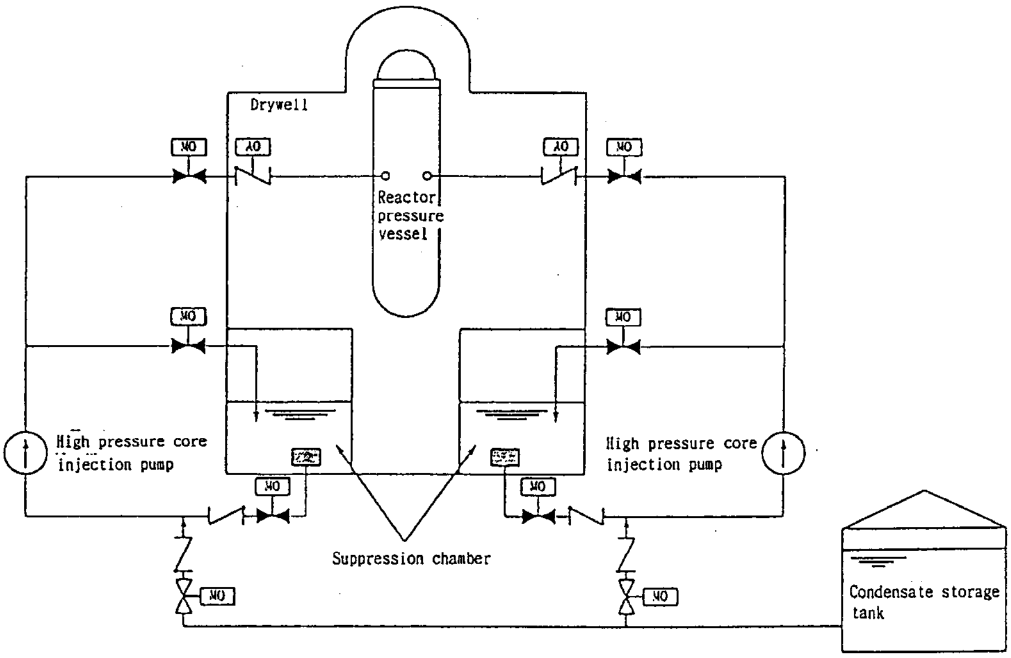


Fig.4.3 Schematic Flow Diagram of High Pressure Core Isolation System

5. Reactor containment facilities

5.1 Purpose of installing reactor containment facilities

The reactor contaonment facilities are provided, as the 4th barrier serving to confine radioactivity, for the purpose of restraining any leakage into the environment to the permissible utmost minimum level, especially for realizing the concept of defense in-depth, even if any fission products (FP) should be emitted from the core. To achieve this purpose, the reactor containment facilities consist of the primary containment vessel (PCV), the PCV spray system for cooling of internal atmosphere and removing radioactive iodine in the vessel, the combustible gas density control system to prevent combustion of gas generated from water-metal reaction in the PCV, the reactor building (secondary containment facility) to functionally prevent any leakage of fission products from direct outgoing into the environment and the stand-by gas treatment system (SGTS) designed to remove FP by filters for overhead emission through the exhaust stack.

5.2 Primary containment vessel (PCV)

The primary containment vessel for the ABWR is made of reinforced concrete with steel liner, inside consisting of the cylindrical dry well enclosing a reactor pressure vessel, the suppression chamber of cylindrical form and the basic slab floor. It contains the diaphragm floor of RC structure separating the dry well from the suppression chamber and the reactor pressure vessel foundation made of steel, in which the horizontal steel vent pipes are provided for connection to the dry well discharge pressure suppression chamber. The primary containment vessel has the vacuum breaker, penetration through the primary containment vessel wall and isolation valves.

In case of any loss of coolant, the mixed emission of steam and water into the dry well can be led into the pool of the suppression chamber, where steam is then cooled and condensed by water in the pool. The design requirement for PCV is to keep down any possible rise of dry well internal pressure and thereby to retain emission of radioactive materials inside the containment vessel. Therefore, the containment vessel is such designed and fabricated as to fully withstand both pressure and temperature conditions at any loss-of-coolant accident. The leak rate for the containment vessel is to be limited to 0.4 percent/d of its spatial volume at the pressure and air conditions equivalent to 0.9 time of the ordinary temperature and maximum working pressure.

5.3 PCV spray system

The primary containment vessel spray system constitutes two independent subsystems. Either one of them, coupled with the LPFL, can be operated to prevent any temperature and pressure rises beyond the designed limits for the containment vessel by removal of out flowing energy and decay heat of coolant resulting from rupture of the feed water pipeline, etc. It further serves effectively to clean out radioactive iodine being suspended inside the vessel. This system represents a mode of the residual heat removal system, which is designed to start up automatically as the LPFL in case of the loss-of-coolant accident and then to work functionally as the spray cooling system by remote-manual switch-over of the motor-operated valve.

The system takes water from the suppression chamber pool for spraying into the containment vessel after cooling of such water by means of the heat exchanger.

Sprayed water is returned to the suppression chamber through the vent pipes.

5.4 Flammable gas control system

The system is designed to prevent combustion of hydrogen and oxygen being generated in the containment vessel at the time of loss of coolant accident. After recombination of hydrogen and oxygen, remaining air goes back to the dry well through the vacuum breaker, so that gas concentration can be controlled in this manner throughout the containment vessel. In many cases, this system is composed of the portable recombiners of full 100 percent capacity and some others. At Unit No.6 and No.7 respectively of Kashiwazaki Kariwa Power Station, the system is arranged for common use at each reactor building.

5.5 Stand-by gas treatment system (SGTS)

The system is composed of exhaust fans, charcoal filters for iodine removal, particle filters of high performance and dehumidifiers. It can start up automatically in case of any such emergency as loss of coolant, etc. and maintain the indoor atmosphere of the reactor building at a negative pressure to check and restrain any emission of radioactivity into the environment.

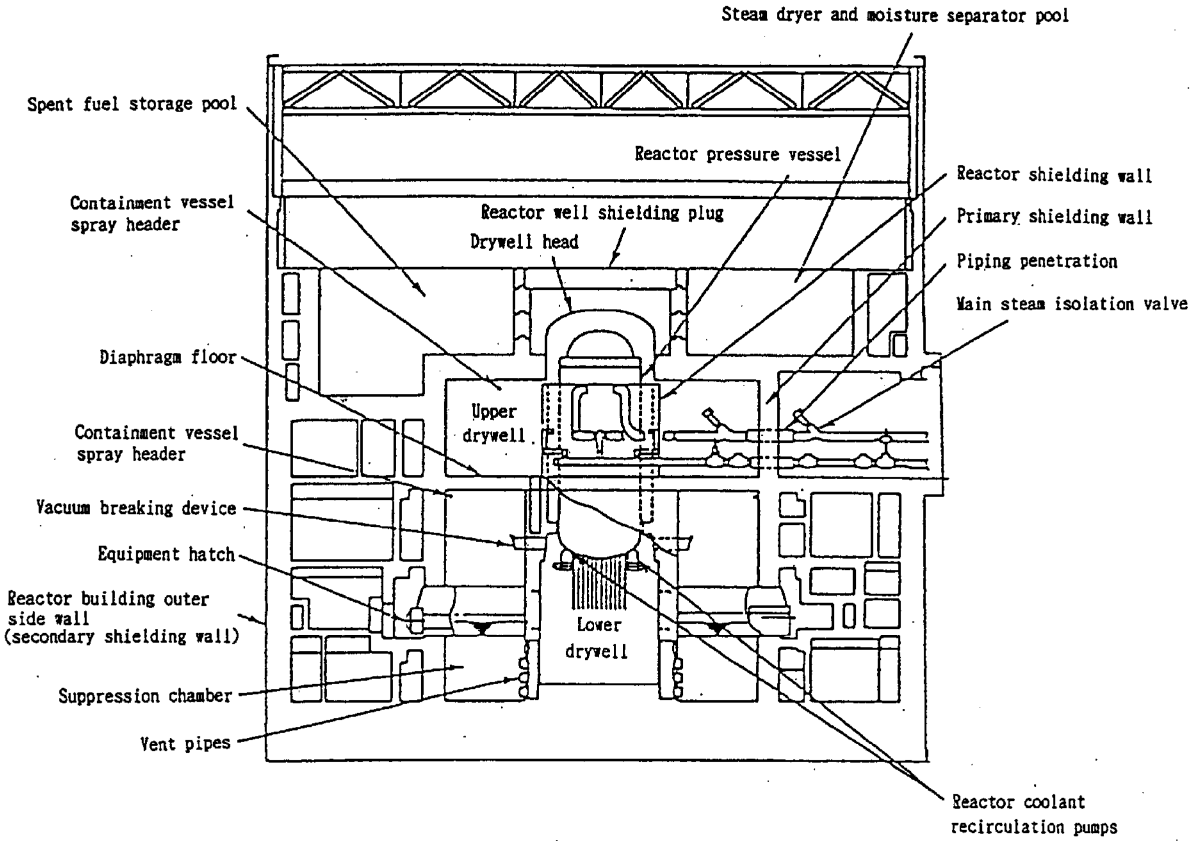


Fig.5.1 Schematic Structure of Reactor Containment Facility (ABWR)

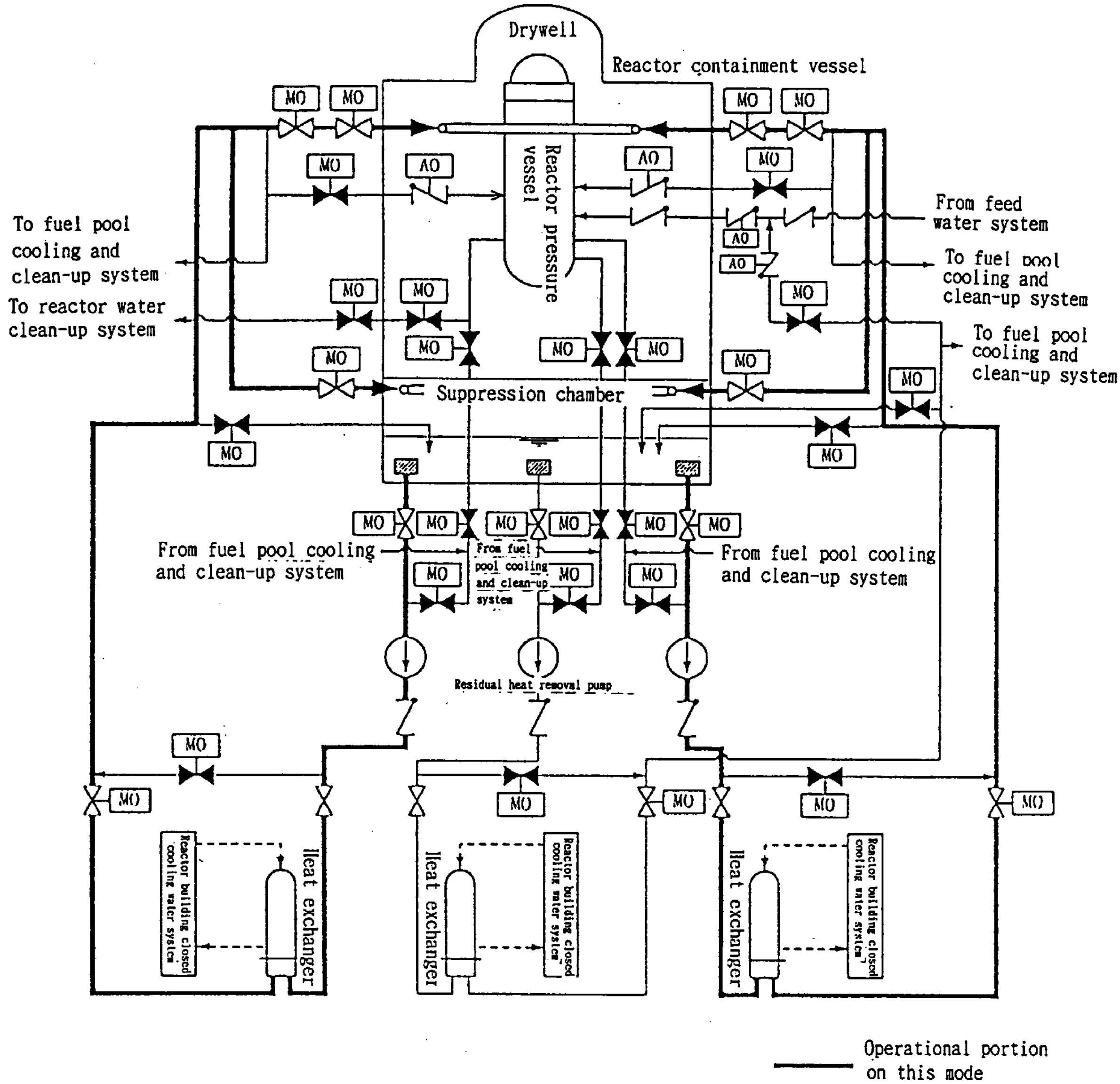


Fig.5.2 Containment Vessel Spray System (RHR system at containment vessel spray cooling mode operation)

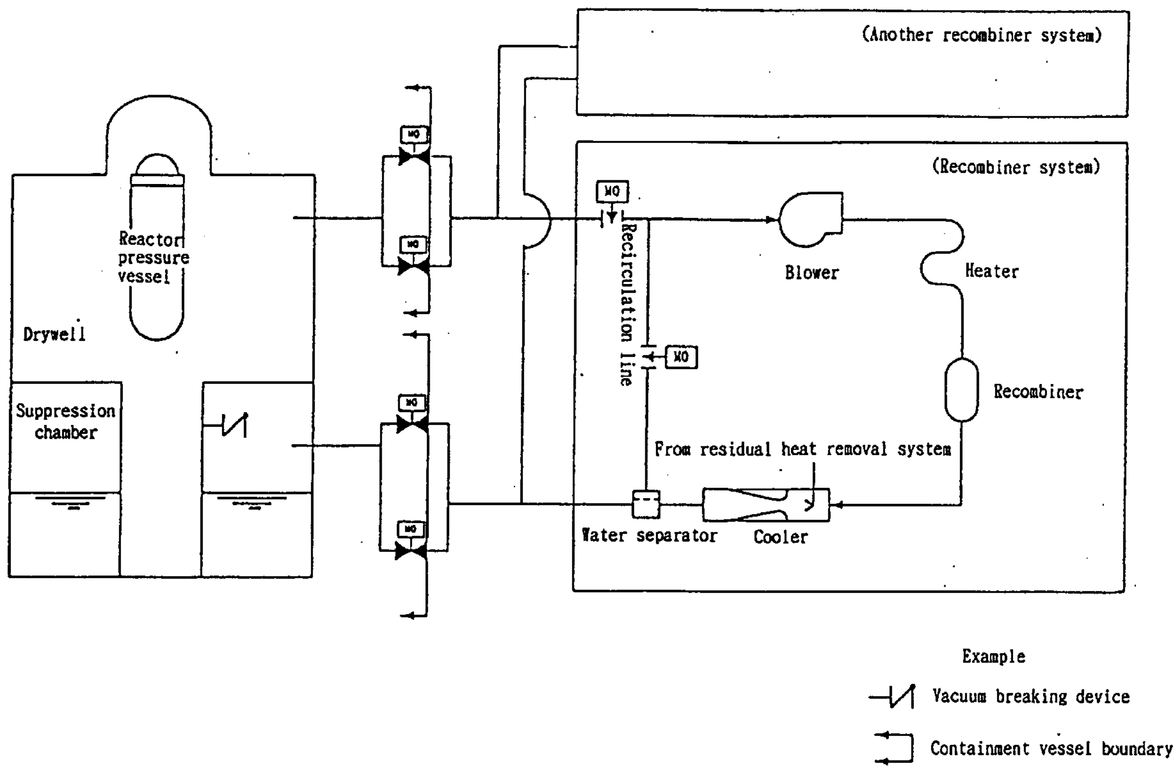


Fig.5.3 Schematic Flow Diagram of Flammability Control System

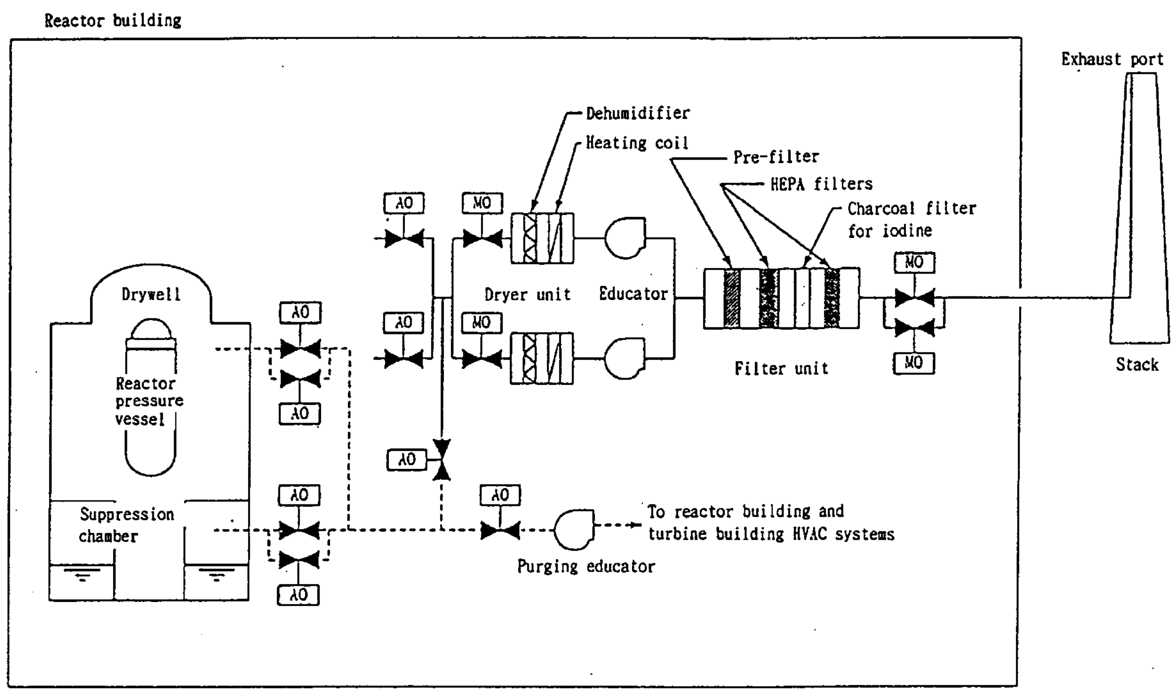


Fig.5.4 Schematic Flow Diagram of Emergency Gas Treatment System

6. Auxiliary systems for reactor

The auxiliary systems for removal of heat from the reactor include the following system:

6.1 Residual heat removal system (RHR)

The RHR system in the ABWR is composed of three (3) loop to be operated for removal of residual heats, such as core decay heat and coolant retention heat, after shutdown of the reactor. It is so designed as to meet the functional needs of multiple operating modes with one single system:

(1) Cooling mode at reactor shutdown

The cooling mode during shutdown is the most common operating mode used for removal of decay heat after shutdown of the reactor. It is so designed as to be able to cool down the reactor water temperature to 52℃ in 20 hours.

(2) Low pressure injection mode and containment vessel spray cooling mode

The residual heat removal system is used, in case of any loss-of coolant accident, as the low pressure flooder system (LPFI) for emergency core cooling and as PCV spray cooling System, while it is in the stand-by condition during normal output operation.

(3) Suppression chamber pool water cooling mode

This mode is designed functionally to take water from the pool, pressurize the water by pumps, cool down water temperature by heat exchanges and return it to the suppression chamber, whenever the water temperature of the suppression pool has risen upon delivery of steam into the suppression chamber by actuation of the main steam relief valve or the reactor isolation cooling system.

(4) Fuel pool cooling mode

This is the mode to be used for cooling the water in the fuel pool in addition to the fuel cooling system, if and when necessary. It serves as a back-up system for fuel pool cooling. if the heat temperature has gone upward beyond the capacity of the fuel pool cooling and purification system, which is provided exclusively for cooling purpose, for instance, as is in the case where the total core fuel has been transferred into the spent fuel pool.

6.2 Reactor core isolation cooling system (RCIC)

The RCIC system is provided for the purpose of supplying fuel cooling water into the reactor, without interruption even when the reactor has been isolated in case of the transient event. When the reactor has been isolated from the turbine facilities and the feed water system has failed to perform its function, steam will be discharged from the mainstream relief valves into the suppression chamber. If this should happen, the reactor core may be exposed after decrease of reactor water level. This is, therefor, just the time for the RCIC system to be put into operation at 'reactor water level abnormal low', thus feeding water from the condensate storage tank into the reactor. Since this RCIC system is so designed as be able to drive the turbine and the pump with use of generating steam, water feed into the reactor is still available even at the time of total power failure. In the design of ABWR, the RCIC system is operated as one of the LPFLs in the emergency core cooling system at any loss-of-coolant accident.

6.3 Reactor auxiliary cooling water system (RCW)

The RCW system is provided for removal of heat being generated in the components of both safety and non-safety systems and also for transfer of residual heat after emergency shutdown of the reactor into the sea as the ultimate heat sink. The system comprises three (3) subsystems corresponding respectively to each of the 3-division emergency cooling systems. Each of them can be operated for cooling independently without any loss of safety function of the total system, even on assumption of a single failure of components.

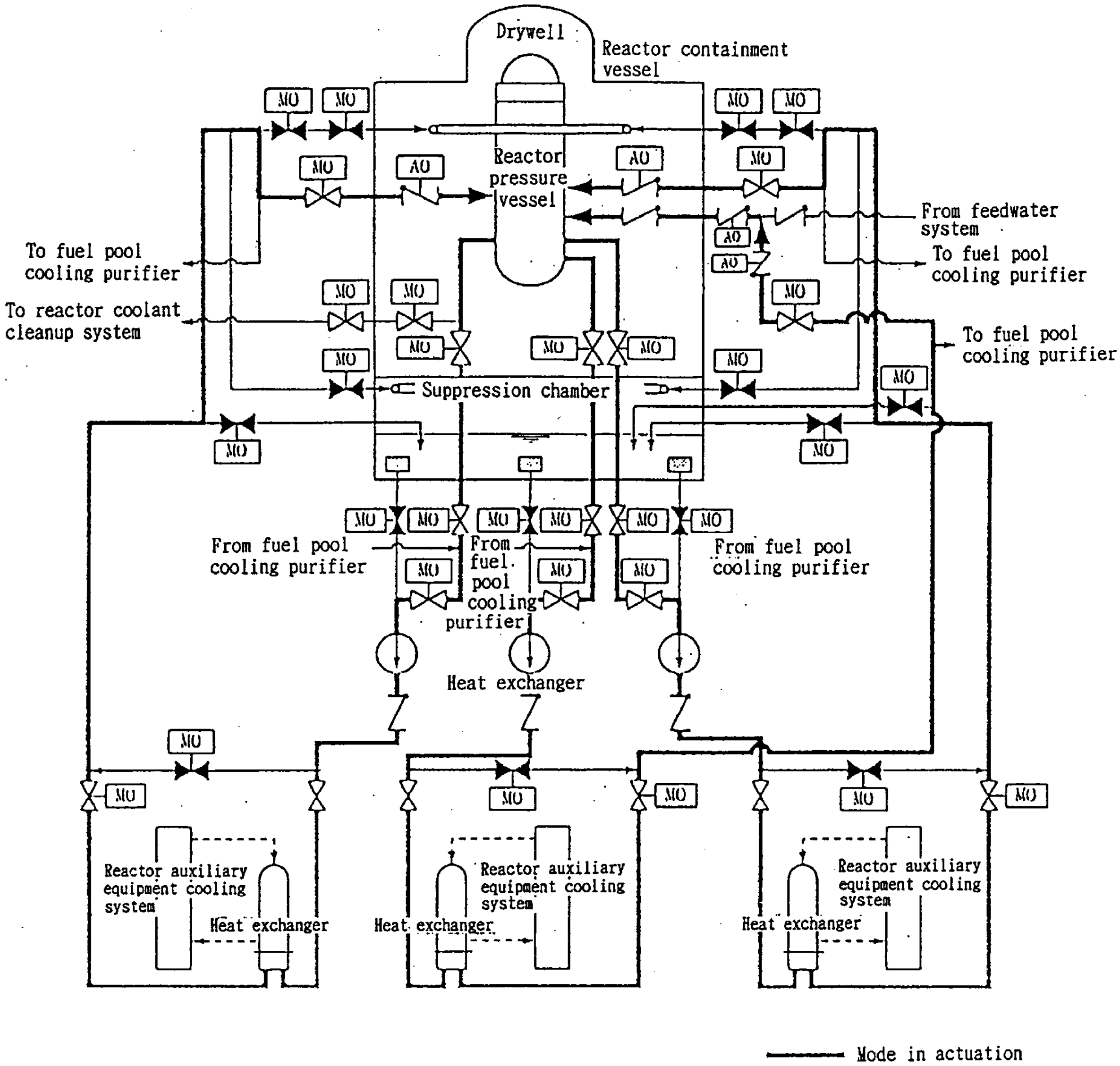


Fig.6.1 Residual Heat Removal System (Cooling Mode at Reactor Shutdown)

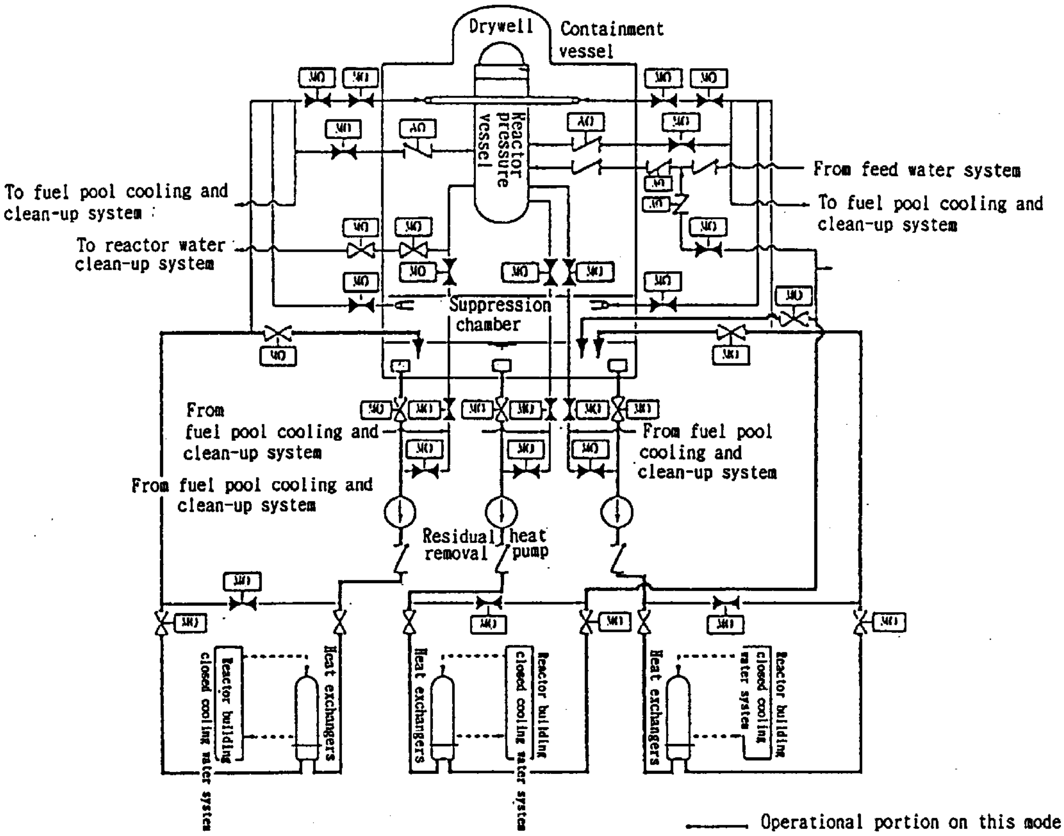


Fig.6.2 Schematic Flow Diagram of RHR System at Suppression Pool Cooling mode operation

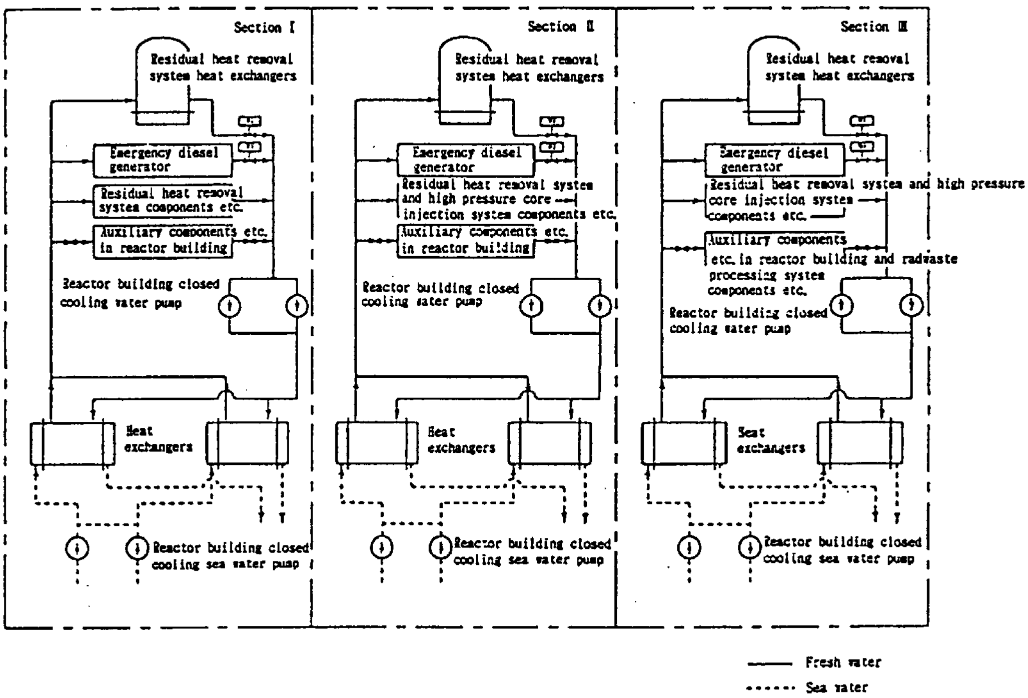


Fig.6.3 Schematic Flow Diagram of Reactor Building Closed Coolant Water System

7. Safety protection system

The safety protection system is provided for protection of the reactor when an abnormal transient is anticipated to occur, or, as a Defense in Depth Level No.2, to prevent further progress of the amnormal condition, which might otherwise endanger the safety of the reactor. The reactor protection system functions also, as Defense in Depth level-3, for activation of safety protection operation in order to mitigate the accident.

The protection system may generally be divided into the two categories; the one devised for emergency shutdown of the reactor in operation and the other for operation of the engineered safety system like the emergency core cooling system.

The safety protection system must be designed according to the following requirements:

a. The protection system should be designed with redundancy, being independent both physically and electrically. The primary function of safety protection should not be impaired or lost against any possible fault or failure on the single unit of equipment or when a single unit of equipment is left out of its operating condition.

b. The Protection system should be designed as to be in its permissible conditions for safety (such as 'fail safe or 'fail as is') even in the event of system power failure or loss of power source.

c. The safety protection system should be completely separated, where applicable, from the non-safety instrument control system. Even if they are partially combined together, there must be no risk of interaction between them, so that the safety protection system can be free from any failure which might take place on the non-safety instruction control system.

d. The safety protection system should be allowed to undergo periodic tests even during plant operation.

e. The safety protection system should be designed with aseismic consideration.

7.1 Reactor shutdown system (emergency shut down system)

In case of the earlier type reactor, (BWR-5, and earlier) all the reactor shutdown system constitute their circuits on the logic basis of (1 out of 2)× 2, while the corresponding system for the ABWR has four (4) separate system by introduction of the 2 out of 4 logic. While the conventional practice was on the basis of the analog instruments or relay sequences logic, the ABWR has its trip channel and logic circuit constituted by the digital system using the microprocessor.

7.2 Operating system for engineered safety system

The engineered safety system is basically activated by the reactor protection system upon detection of abnormal decrease of reactor water level or unusual increase of dry well pressure. Such events may arise from loss of coolant due to any damage on the primary system of the reactor. The low pressure core cooling system (LPFL-an operating mode of the residual heat removal system) can perform their role of water injection into the reactor core, only if and when the reactor pressure has been reduced to a lower level, because of the low discharge pressure of the pumps in operation. The automatic depressurization system can start up to reduce pressure of the reactor. when the AND-logic of 3 signals with reactor water level low, dry well pressure high and low pressure flooder in operation is satisfied.

In addition to that, the stand-by gas treatment system will be activated to maintain a negative pressure inside the reactor building, so that any possible radioactive leakage into the environment can be prevented. In order to ensure power source available for emergency, the diesel generators for ECCS must be put into service.

The operating system for engineered safety facilities in the ABWR plant is also based on the digital system by use of the microprocessor and the 2 out of 4 logic for its theoretical background is introduced.

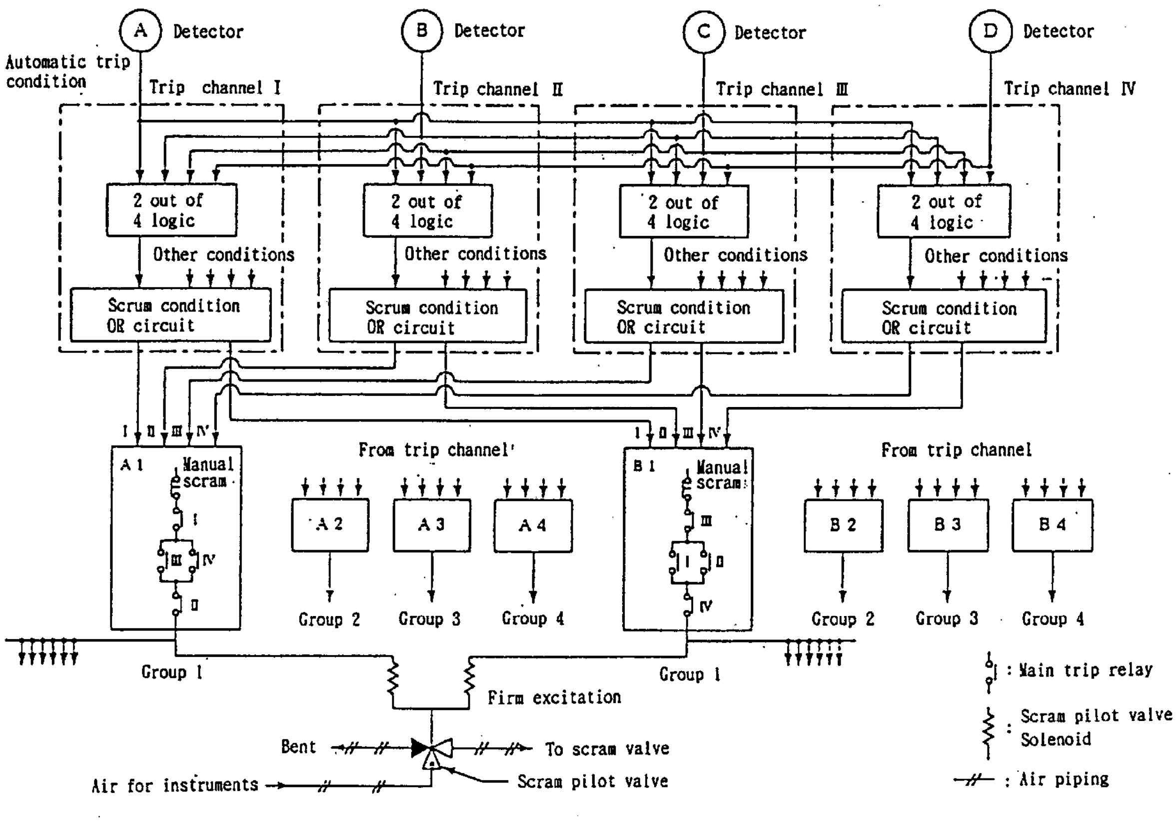
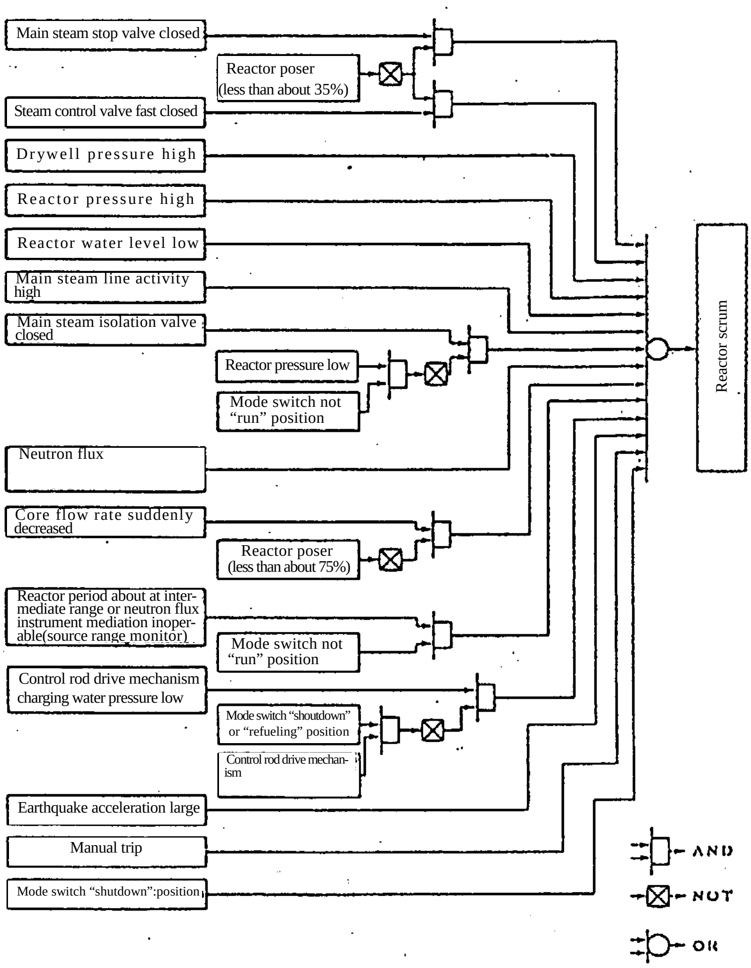


Fig.7.1 Schematic Diagram of Operating Circuit for Reactor Emergency Shutdown System

Fig.7.2 Block Diagram of Reactor Emergency Shutdown System



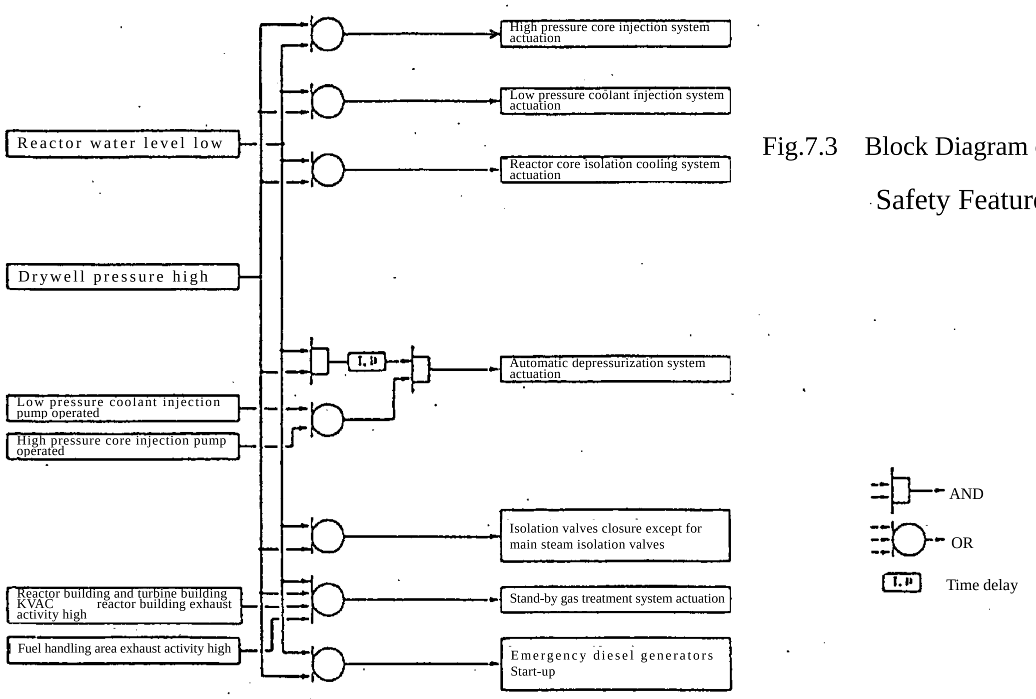


Fig.7.3 Block Diagram of Engineered Safety Feature Actuation System

8. Electrical system

8.1 Outline of design principles in safety-related electrical systems

(1) The on-site power system is provided for transmission of the generated power output to the outside service area and also for adequate power generating source to operate all the necessary auxiliary equipment of the plant for startup, normal operation and emergency operation at loss of coolant accident.

(2) The plant is provided with both off-site power supply and emergency power supply system as may required to secure the safety functions for the major buildings, structures, system and equipment.

(3) The emergency power supply system, as the power source available for all those equipment connected to the safety protection system and the engineered safety system, must be so designed as to be separated from the others electrically and physically.

(4) The design for the electrical system should ensure that, even on assumption of sigle failure, the main functions for safety protection would not be lost under the operating conditions either with the emergency power source or with the off-site power source.

8.2 Stand-by diesel generating set

The stand-by diesel generating set for emergency provides its service as may be necessary for the safe shutdown of the reactor when the external power system has fallen into failure. In case of the loss-of-coolant accident taking place with failure on the off-site power source, it must provides its power to actuate the engineered safety system. In view of required redundancy and independence for this system, the diesel generating system has its triple subsystem for power supply to ensure safe operation of the connected equipment on assumption of a single failure, when the external power has fallen into outage.

Diesel fuel reserved for storage in quantity is required to allow the continuous operation of the diesel generators for about seven (7) days.

8.3 Direct-current power supply system

The DC power supply system provides its power to the DC control circuit in the normal operational condition of the plant and in its emergency condition as well, and the supply is also at outage of the alternating-current power to the emergency power circuit.

This system, consisting of battery, electric charger and DC control center, etc., is designed with the philosophy of redundancy and each one of the redundant and isolated unit is capable of supplying electricity independently to each DC control train of the redundant safety protection systems and engineered safety systems.

8.4 Power supply system for instrument control

This system serves as the AC power supply source for control of plant instrumentation. Power is supplied by selecting an appropriate power source, depending on the required quality of power source and according to the priority of importance with regard to safety and operation of the equipment.

9. Characteristic features of BWR in terms of safety

In supplement to the safety design of BWR as outlined above, the characteristic features of BWR in terms of safety may be summarized in the following:

(1) Self-controllability of power output

The first one of the features is that there is generation of void (steam foaming) due to boiling of coolant at the core during normal operation. This implies that when the output increases for some causes, the degree of boiling water tends to increase accordingly with resultant reaction toward reducing the thermal neutron flux, thus effectively suppressing the rate of power output increases.

(2) Large cooling capability by natural circulation

The second feature is that the BWR has a large capacity of natural circulation which is, in itself, capable of removing core heat at and up to about 50 percent output, even though the recirculation pump has suspended its operation.

(3) Monitoring of water level

The third point of its features lies in the full-time direct monitoring over the reactor water level by means of the level gage. The control system is such designed that the water level can be automatically adjusted by regulation of the feed water flow if and when the level gage indication is varied.

(4) Water pooling in large quantity

The fourth point refers to plenty of water reserve in the pool inside the suppression chamber of the primary containment vessel. Because of this reserve, decay heat to be generated in the core can be absorbed for some time into the pool: even if the circumstance demand complete separation of the reactor. Furthermore, even on assumption of the loss of coolant accident (LOCA) caused by damage on the primary coolant system, the reservoir pool serves to absorb thermal emission and is functionally capable of feeding water into the emergency core cooling system (ECCS) over a long period, thus contributing much toward higher safety.

APPENDIX TO OUTLINE OF SAFETY DESIGN

(CASE OF BWR)

OUTLINE OF SAFETY DESIGN
IN
ABWR

1. Purpose of ABWR Development

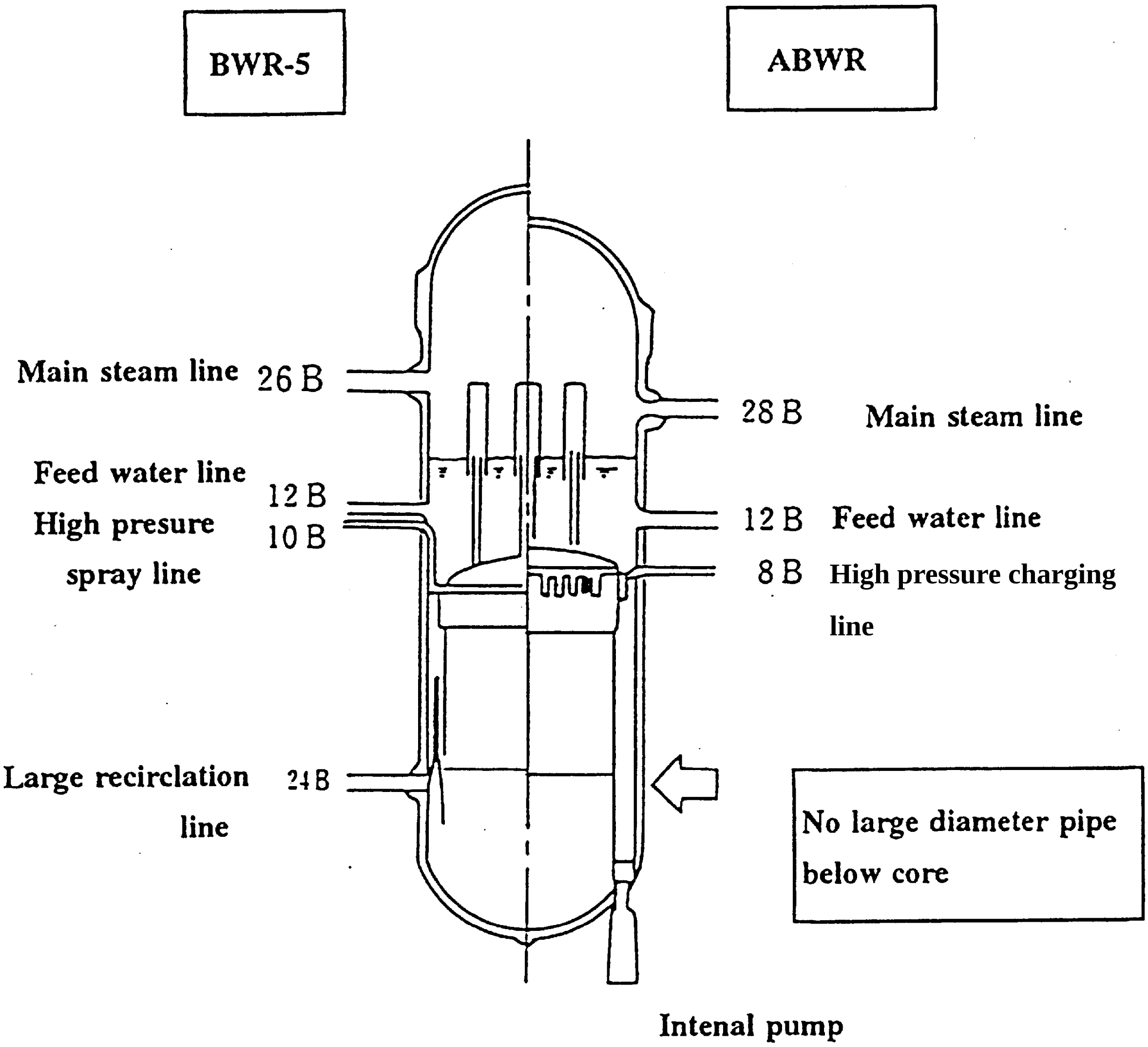
   - Improvement of Safety and Reliability
   - Improvement of Operability and Maneuverability
   - Reduction of Radiation Exposure

2. Reduction of Postulated Pipe Rupture Diameter
   (Internal Pump)
   Elimination of Large Recalculation Piping

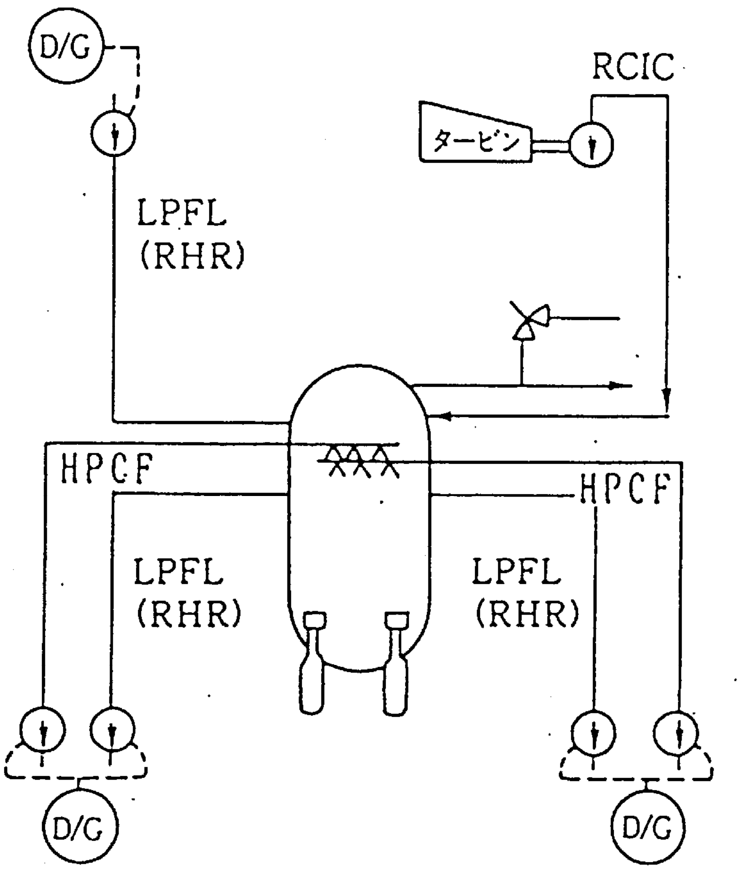
3. Improvement of Reactor Shutdown Capability
   (Fine Motion Control Rod Drive Mechanism)
   Diversity of Driving Source
   Hydrauric Scramming
   Electrical Motor Drive Insertion
   Normal Drive
   Fine Motion by Electrical Motor

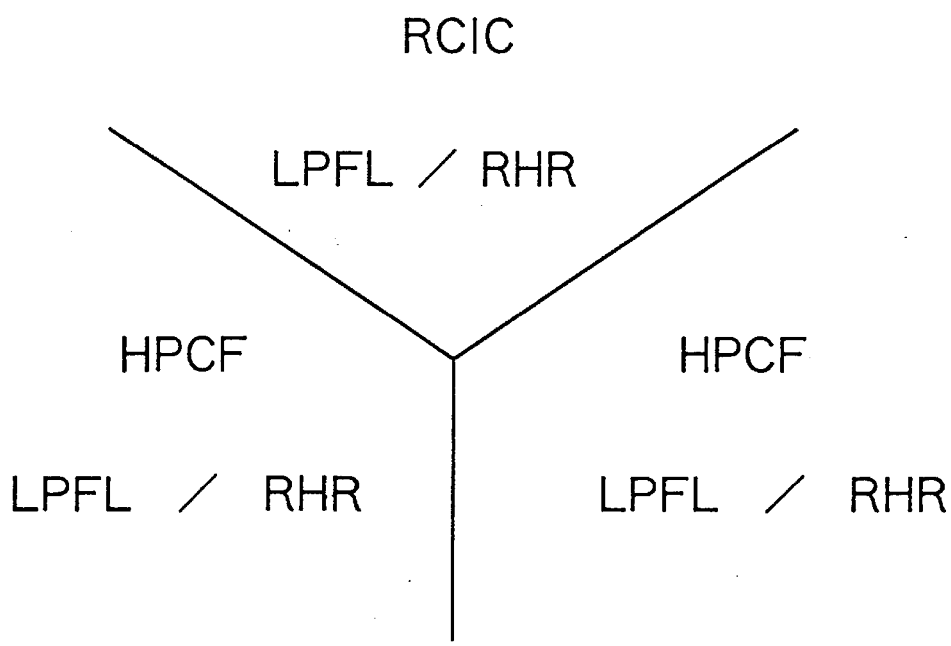
4. ABWR Technical Features and Plant Performance

PERF. / Tech. Feature	Plant Safety	Plant Availability	Plant Operability	Radiation Exposure Reduction	Economic Rationaliza-tion
1. Large Capacity					○
2. Improved Core		○	○		○
3. Internal Pump	○		○	○	
4. Fine Motion CRD	○	○	○	○	
5. Three-division ECCS	○				
6. RCCV	○ (Aseismatica-ly)				
7. New I&C	○	○	○		

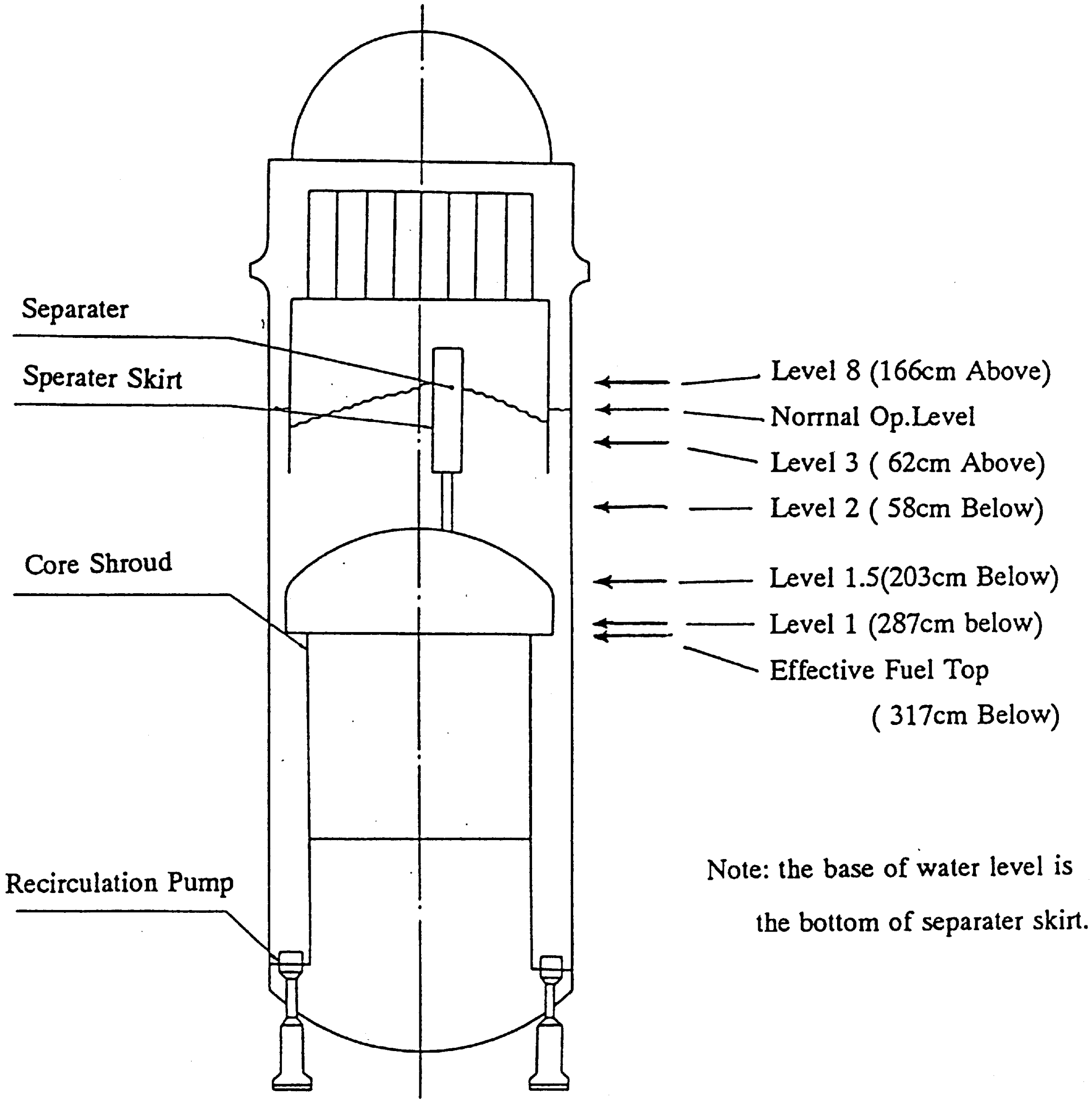


Elimination of Large Recirclation Pipe By Adaptation of Internal Pump





3-Division In ABWR ECCS



Water Level In Reactor Core

Function of Water Level Signal

Level 8:	Turbine Trip
	feed Water Pump Trip
Level 3:	Reactor Scram
	4 Recirculation Pumps Trip
	Stand-by Gas Treating System Start
Level 2:	RCIC Start
Level 1.5:	HPCI Start
	MSIV Closure
	D/G (Div. II III)Start
Level 1:	LPFL Start
	Ads Start
	D/G (Div. I)Start

(OHP PRESENTATION MATERIAL)

OUTLINE OF SAFETY DESIGN

(CASE OF BWR)

Outline of Safety Design
(Case of BWR)

September, 2005

Japan Nuclear Energy Safety Organization

Concept of Safety Design in Nuclear Reactors

- 3 Principle of Reactor Safety
  1. Shutdown

     Reactor shutdown at abnormal condition
  2. Cooling

     Residual heat removal
  3. Contain

     Contain of radioactivity

Multiple Physical Barriers

- 5 Barriers for Containing Radioactivity

  Level-1: Fuel Pellet Matrix

  Level-2: Fuel Rod Cladding

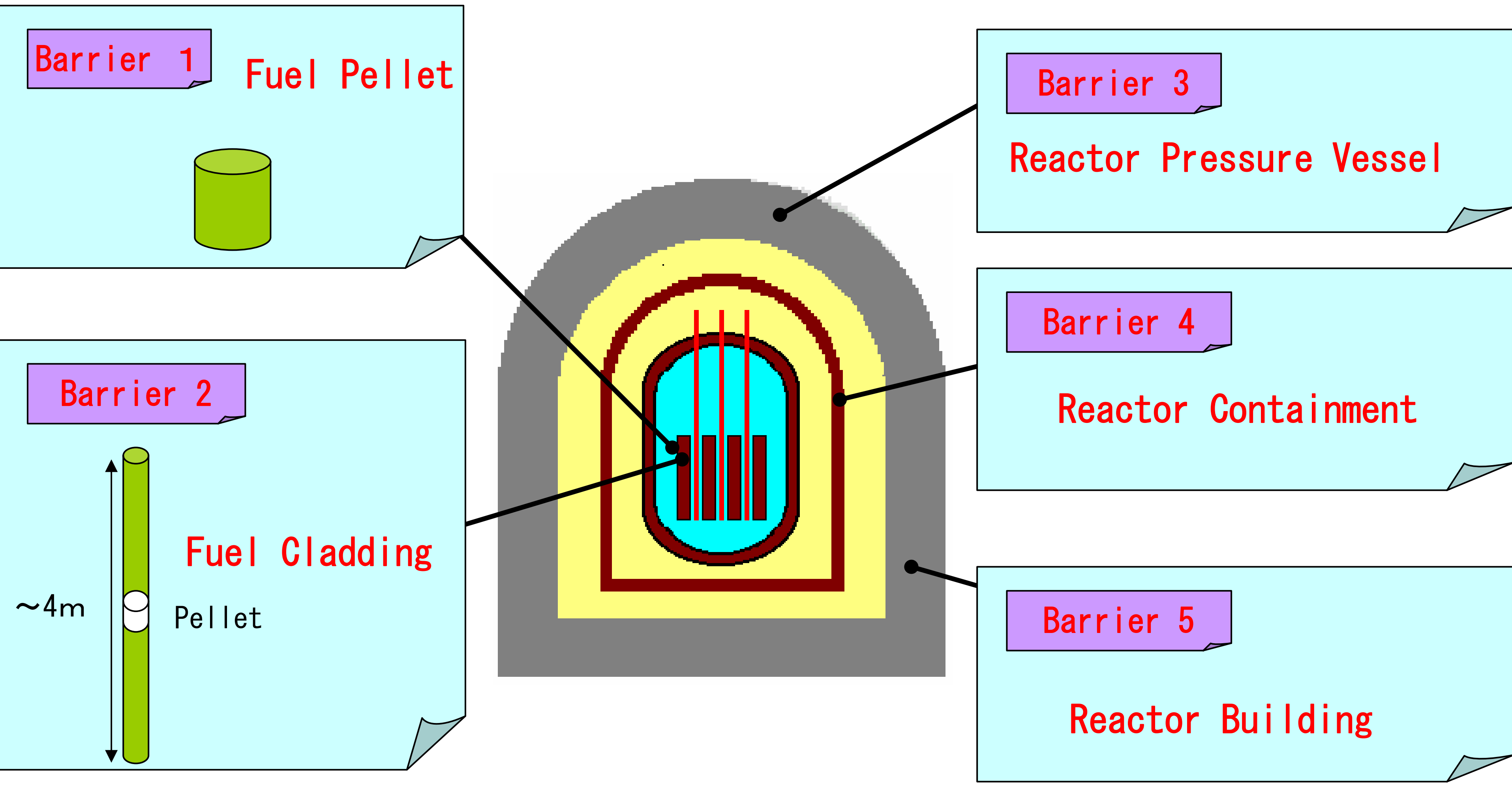
  Level-3: Reactor Primary Coolant Pressure Boundary

  Level-4: Primary Containment Vessel

  Level-5: Secondary Containment Facilities

Protective Barrier for Containing Radioactivity

-Quintuple Barriers-



Defense In Depth

- 3 levels defense

☆First level (Prevention of Abnormal Conditions)

● Design Margin

● Careful Maintenance

● Quality Assurance

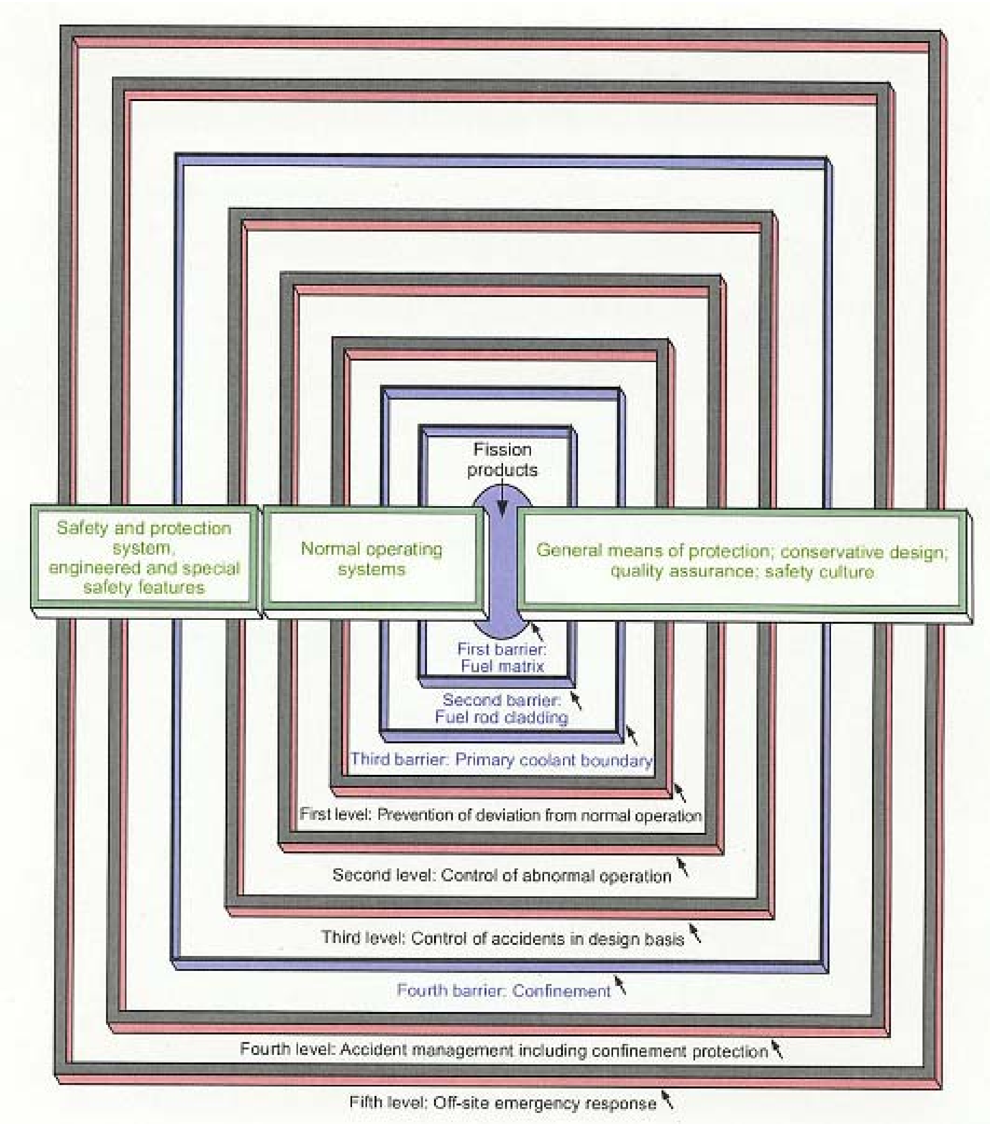
☆Second Level (Prevention of Abnormal Condition propagation)

● Detection of Abnormal Condition (Reactor Shutdown)

☆Third Level (Mitigation of Abnormal Condition)

● Emergency Core Cooling System

● Confinement of Radioactivity in Containment



The relation between physical and levels of protection in defense in depth

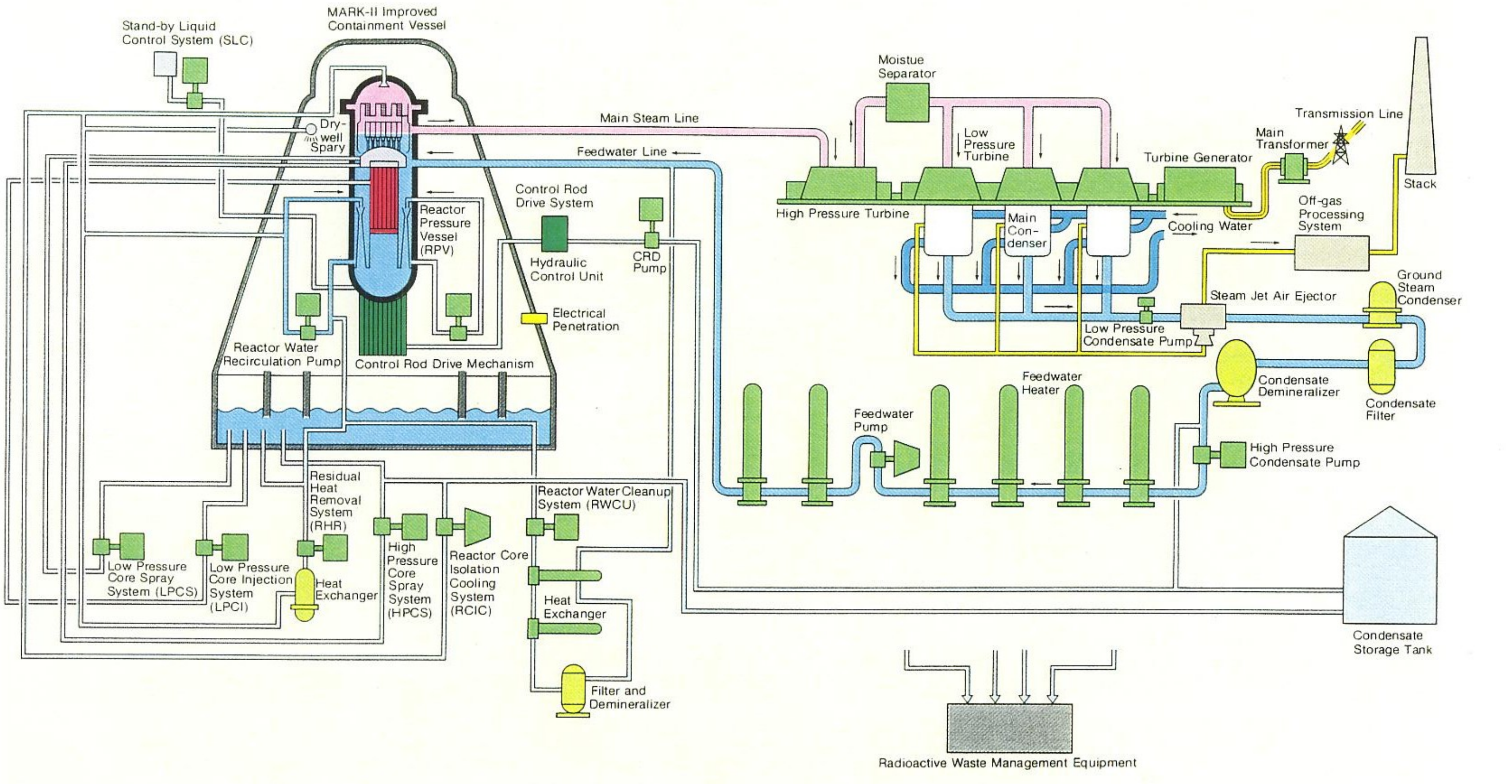
Single Failure Criteria

- Definition

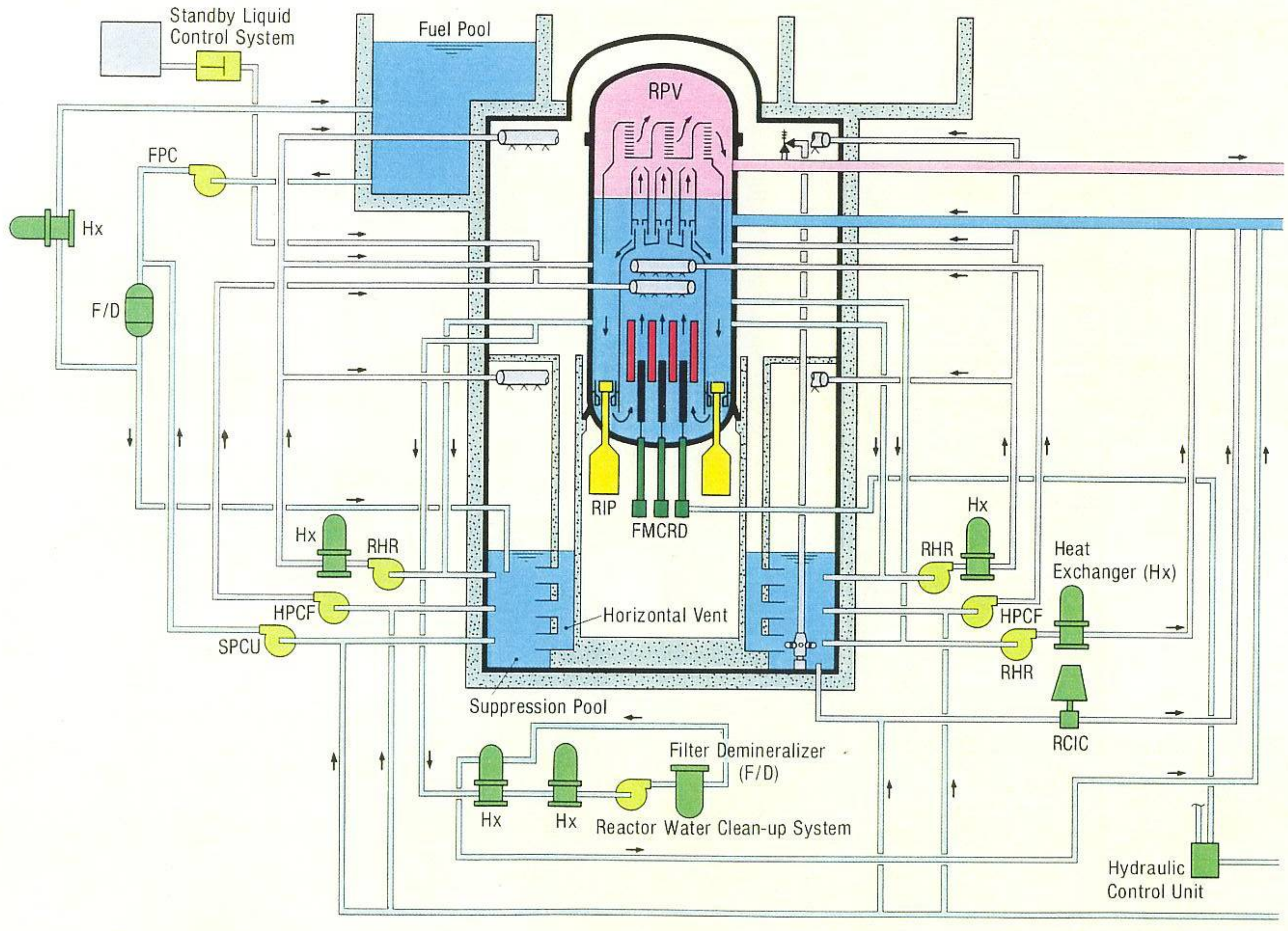
  Single Failure Criteria means that a prescribed safety function of a component is lost due to single event.

  Single Failure Criteria includes inevitable multiple failures due to single event.

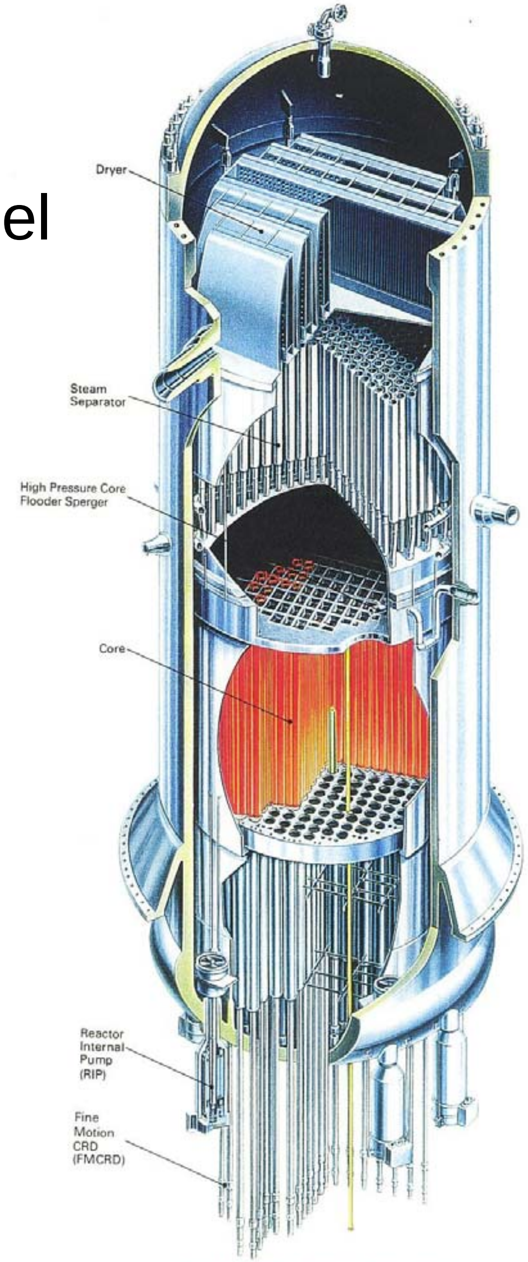
BWR Nuclear Power Plant



ABWR Nuclear Power Plant



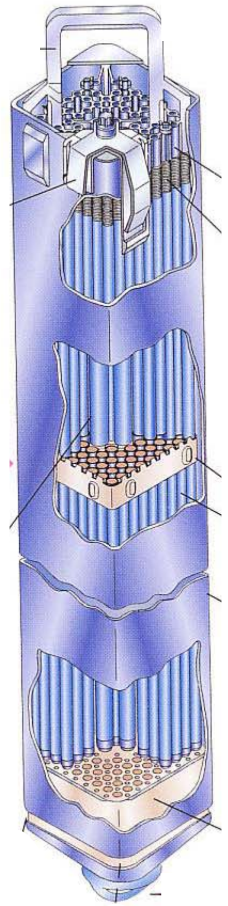
ABWR
Reactor Pressure Vessel



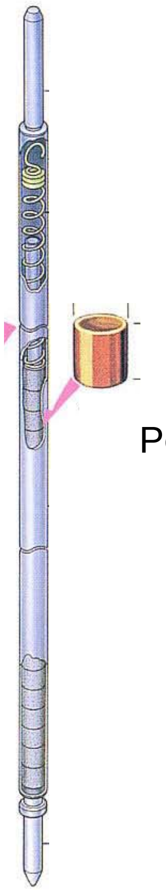
REACTOR PRESSURE VESSEL AND INTERNALS

Fuel Assembly

Fuel assembly

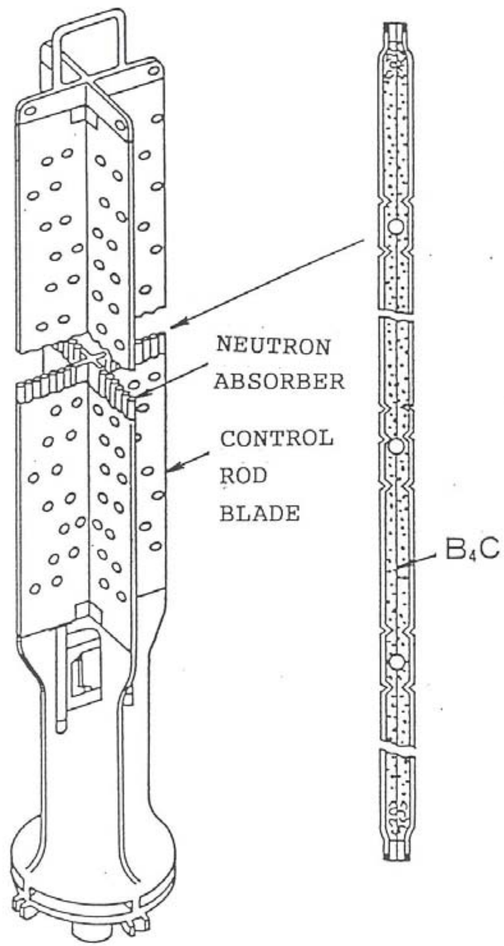


Fuel Rod



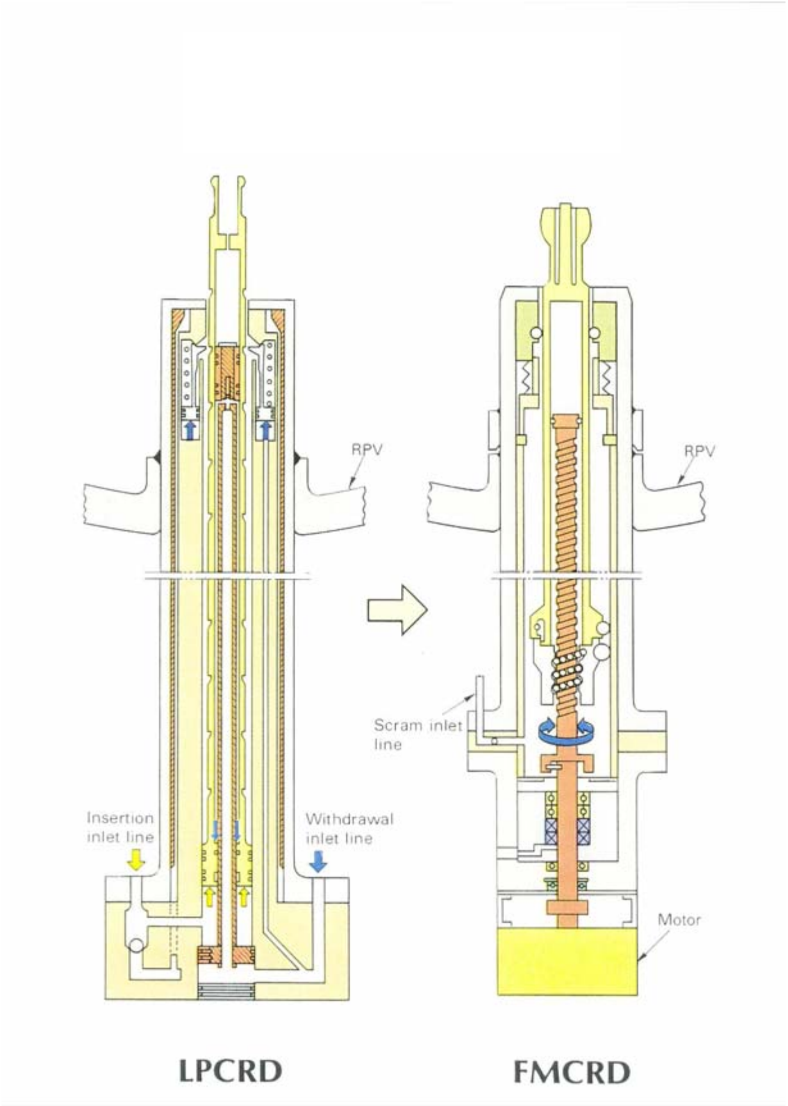
Pellet

Control Rod

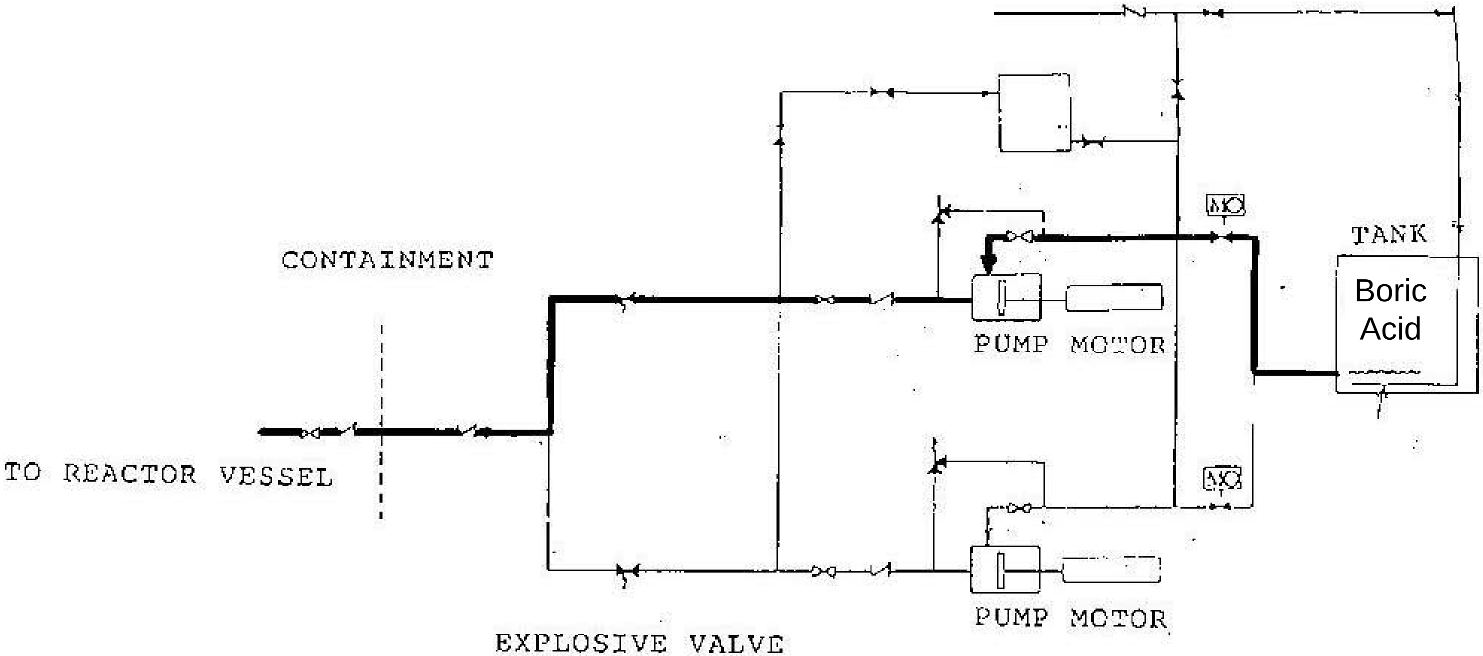


Control Rod

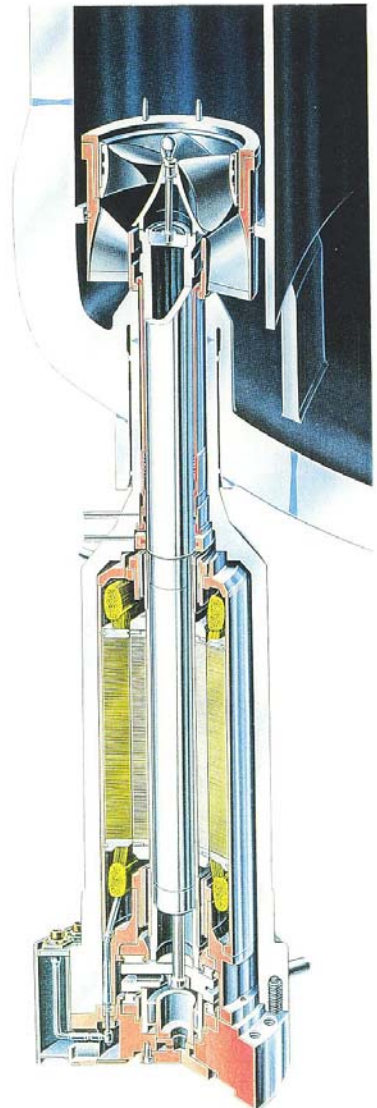
Control Rod Drive System



Stand By Liquid Control System



Reactor Internal Pump



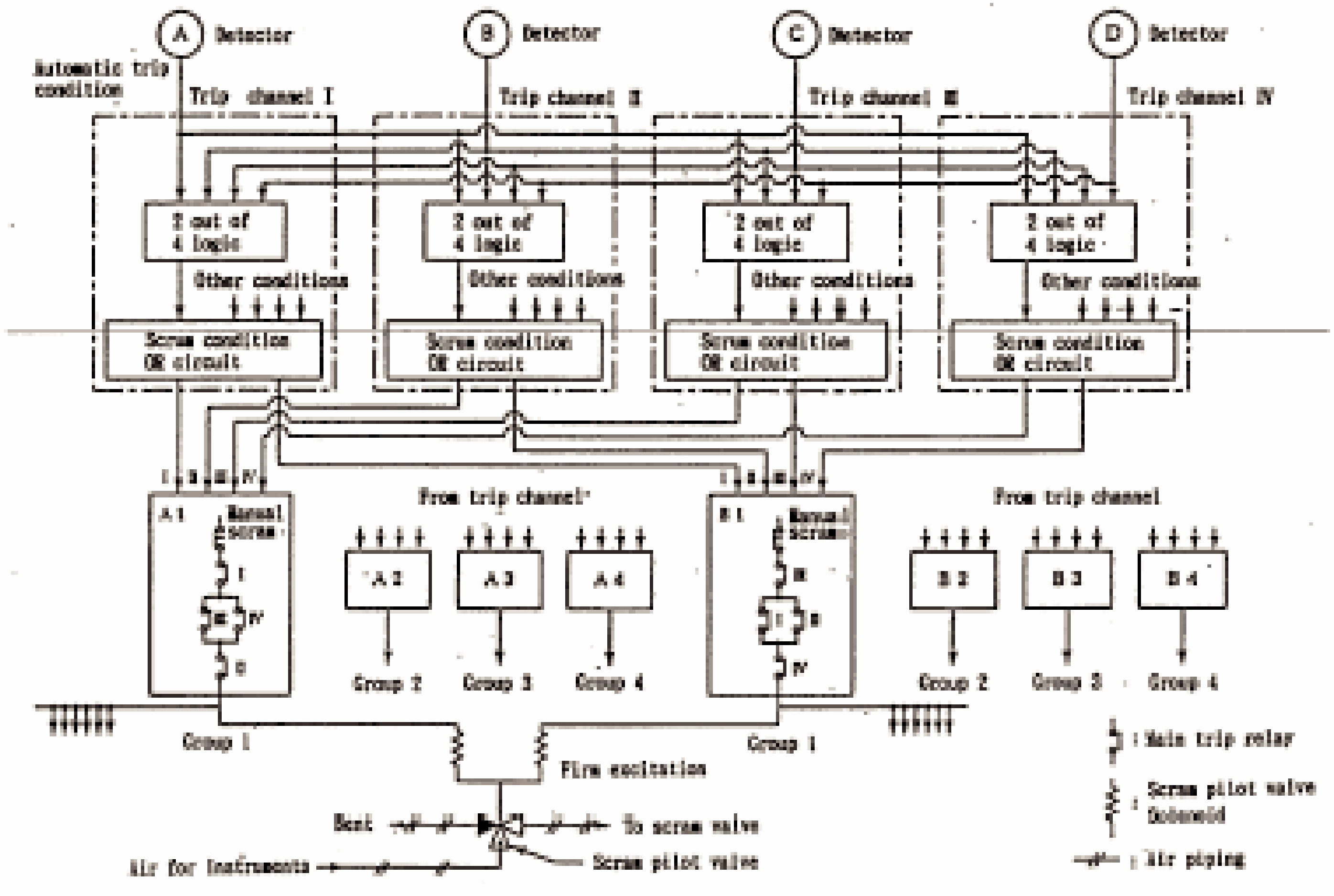
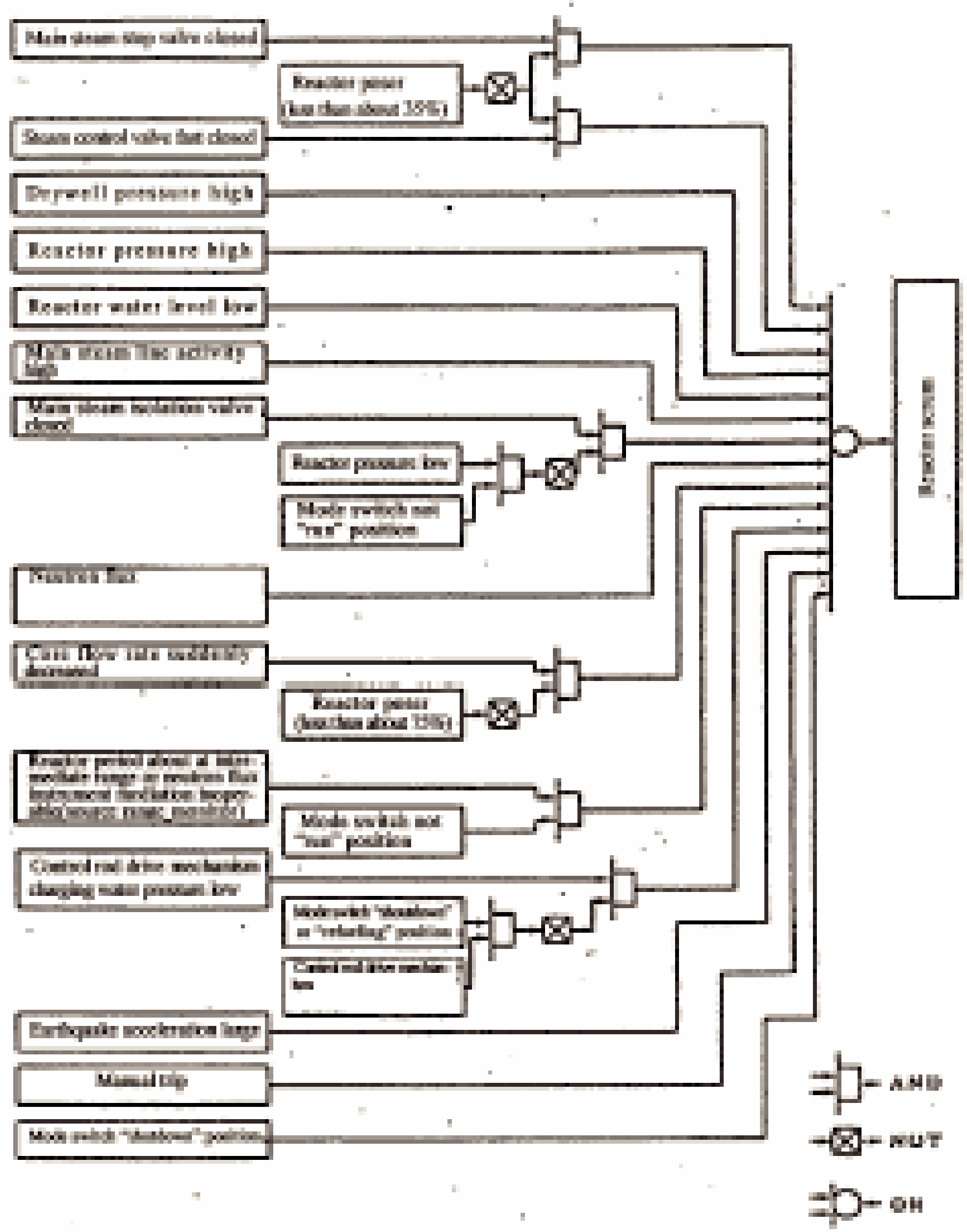


Fig.7.1 Schematic Diagram of Operating Circuit for Reactor Emergency Shutdown System

Fig.7.2 Block Diagram of Reactor Emergency Shutdown System



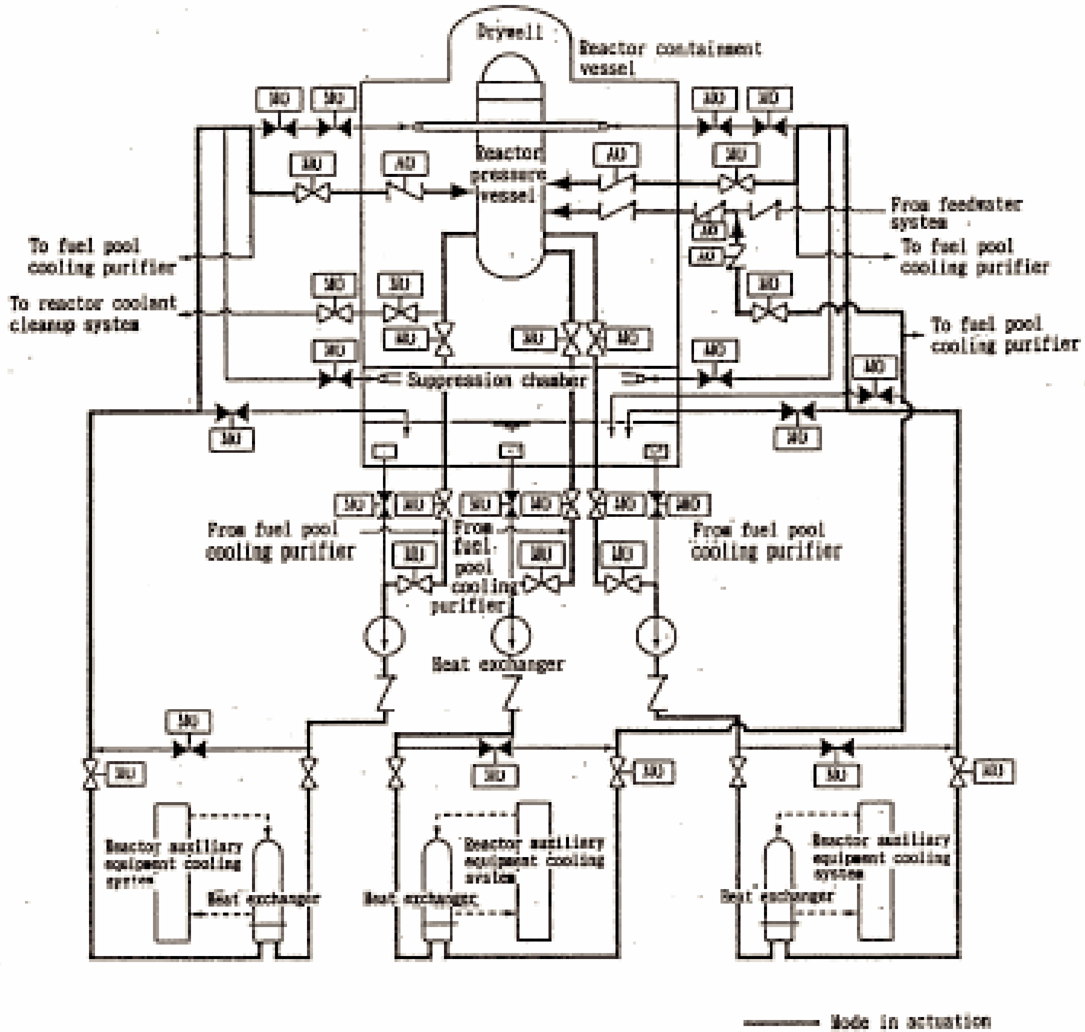


Fig.6.1 Residual Heat Removal System (Cooling Mode at Reactor Shutdown)

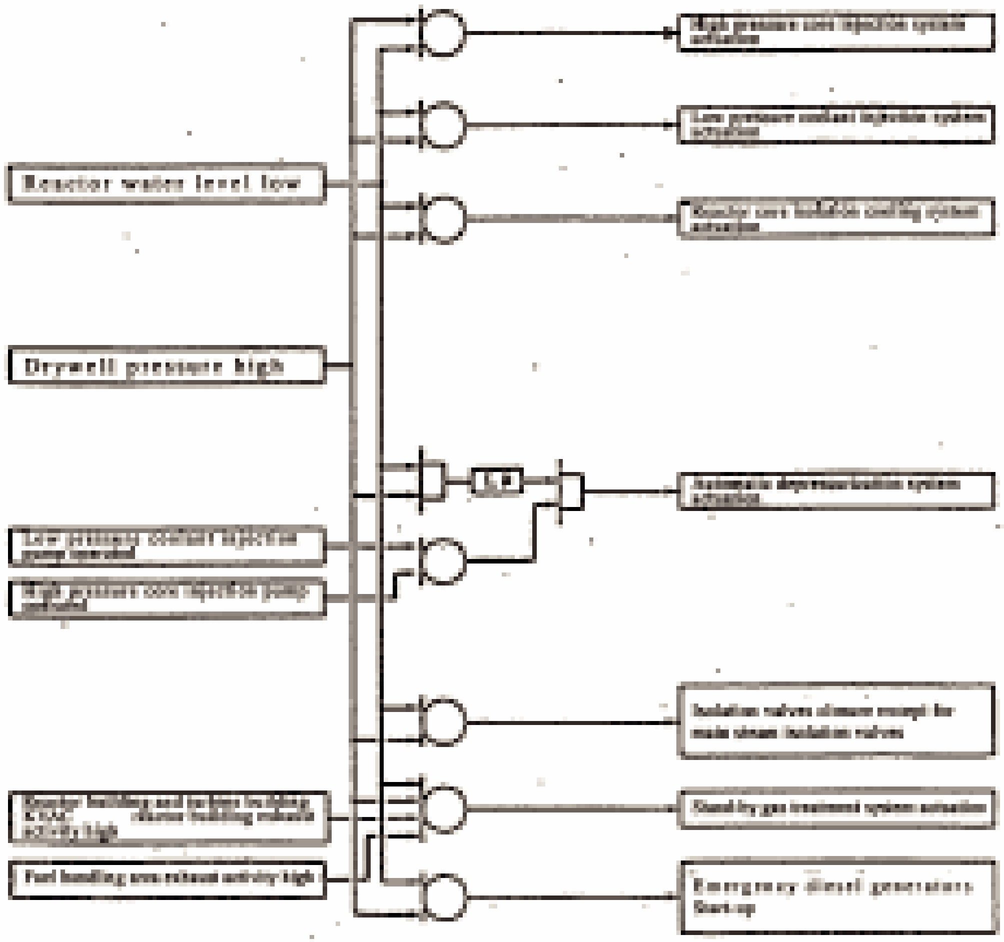
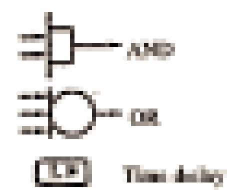
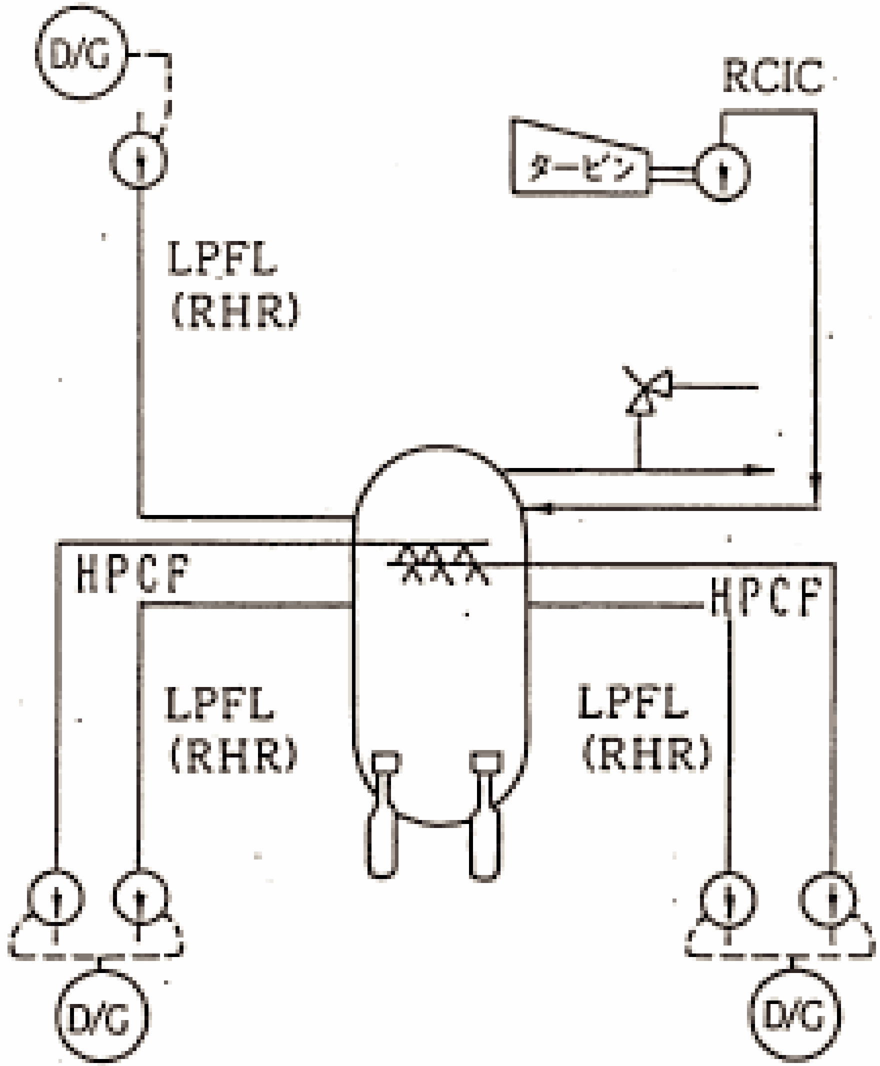


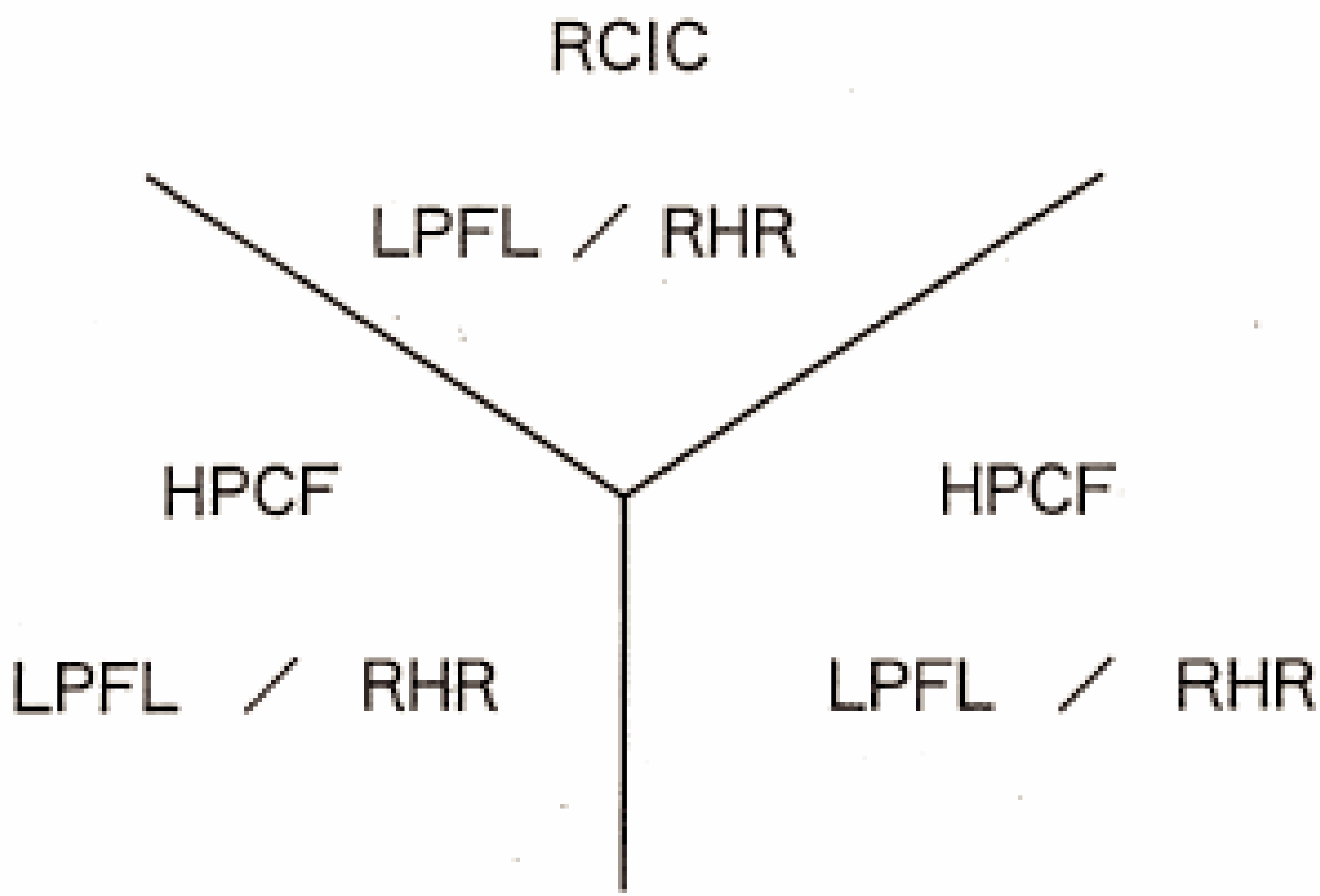
Fig.7.3 Block Diagram of Engineered Safety Feature Actuation System



Number Of Train And Its Capacity In Safety System

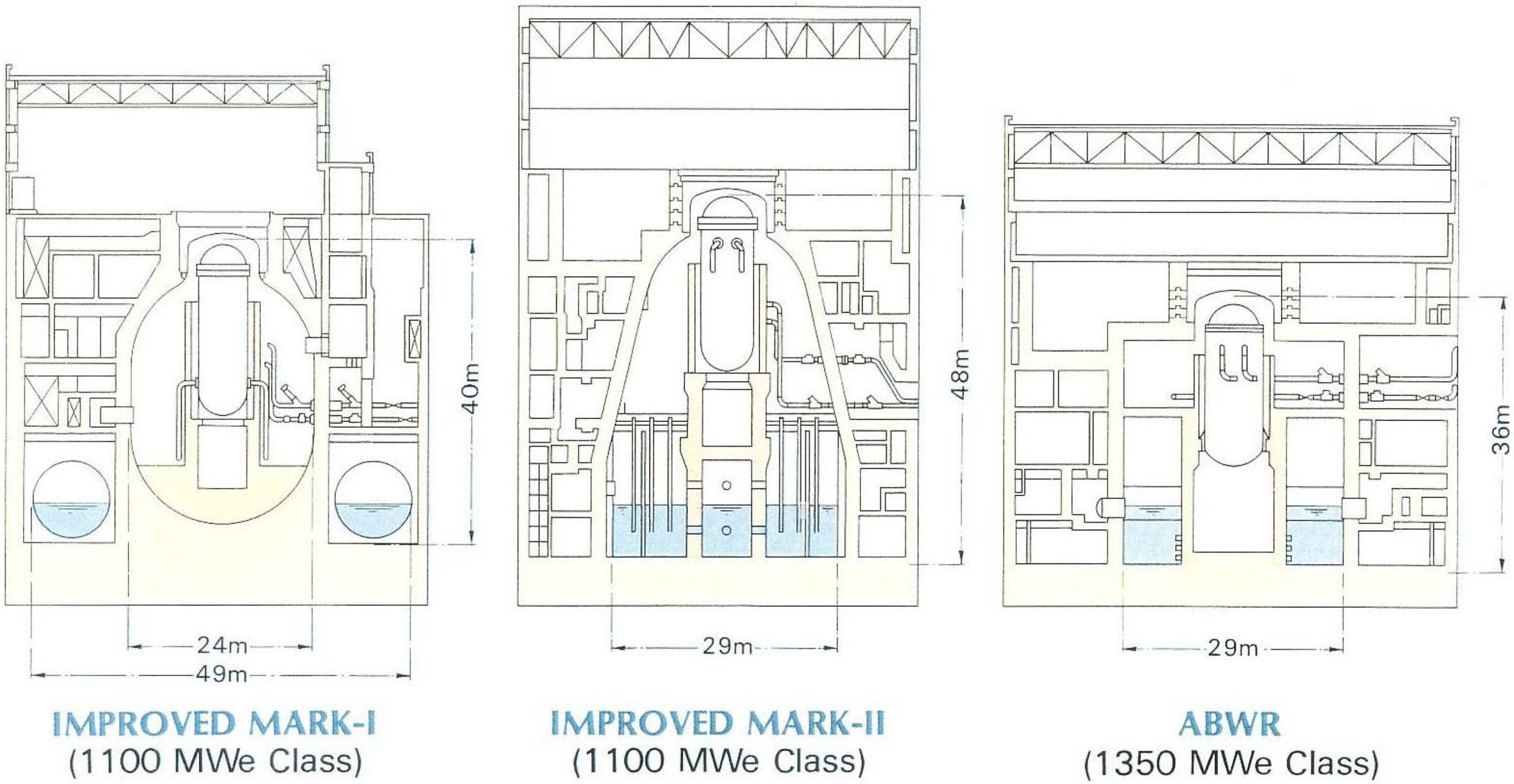
Requirement / Capacity	N	N+1	N+2
	- Necessary For Normal Operation	- Same As Left - Single Failure	- Same As Left - Single Failure - Maintenance
100% for 1 Train	1 Train 100%	2 Train 100% + 100%	3 Train 100% + 100% + 100%
50% for 1 Train	2 Train 50% + 50%	3 Train 50% + 50% + 50%	4 Train 50% + 50% + 50% + 50%
33% for 1 Train	3 Train 33% + 33% + 33%	4 Train 33% + 33% + 33% + 33%	5 Train 33% + 33% + 33% + 33% + 33%


D/G
RCIC
タービン
LPFL
(RHR)
HPCF
HPCF
LPFL
(RHR)
LPFL
(RHR)
D/G
D/G



3-Division In ABWR ECCS

BWR Reactor Containment



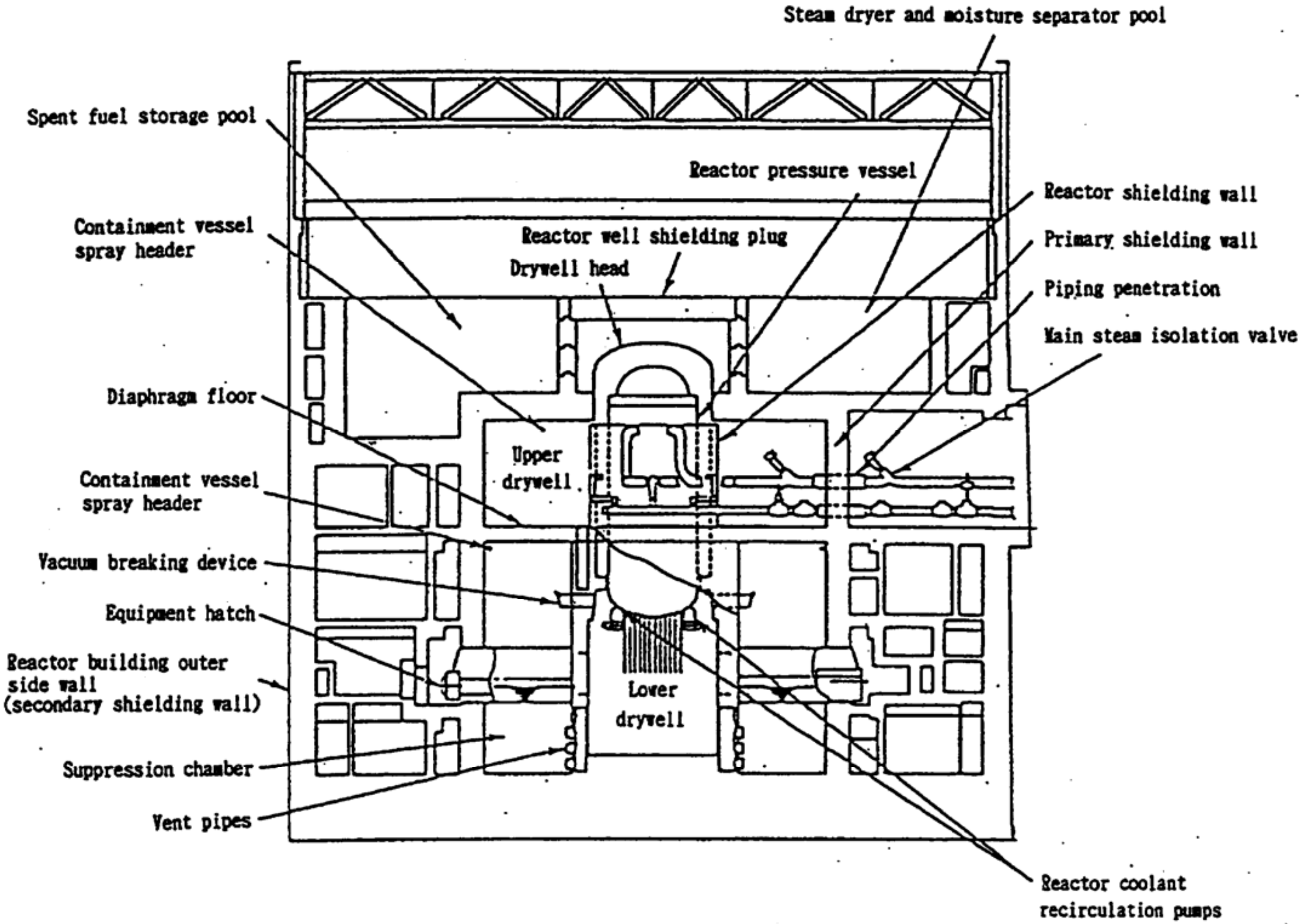


Fig.5.1 Schematic Structure of Reactor Containment Facility (ABWR)

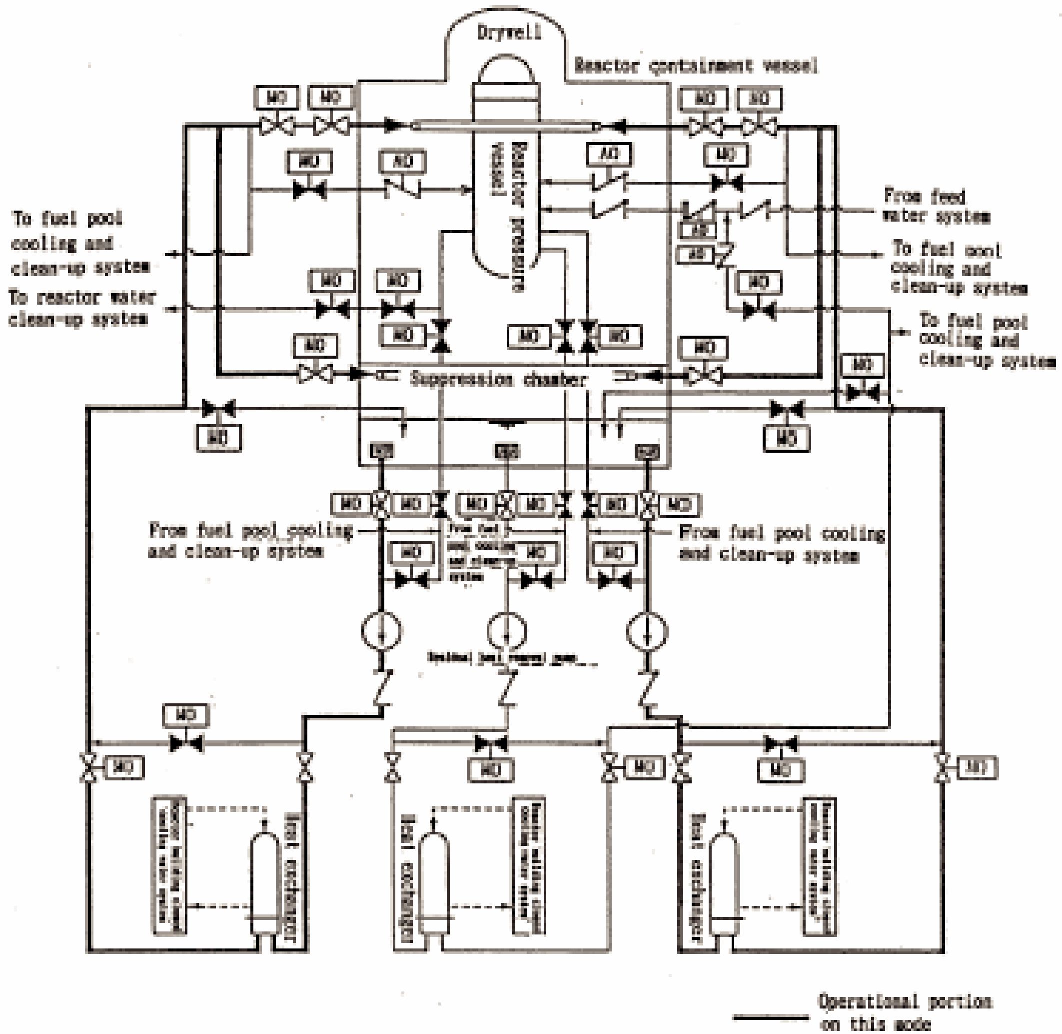


Fig.5.2 Containment Vessel Spray System (RHR system at containment vessel spray cooling mode operation)

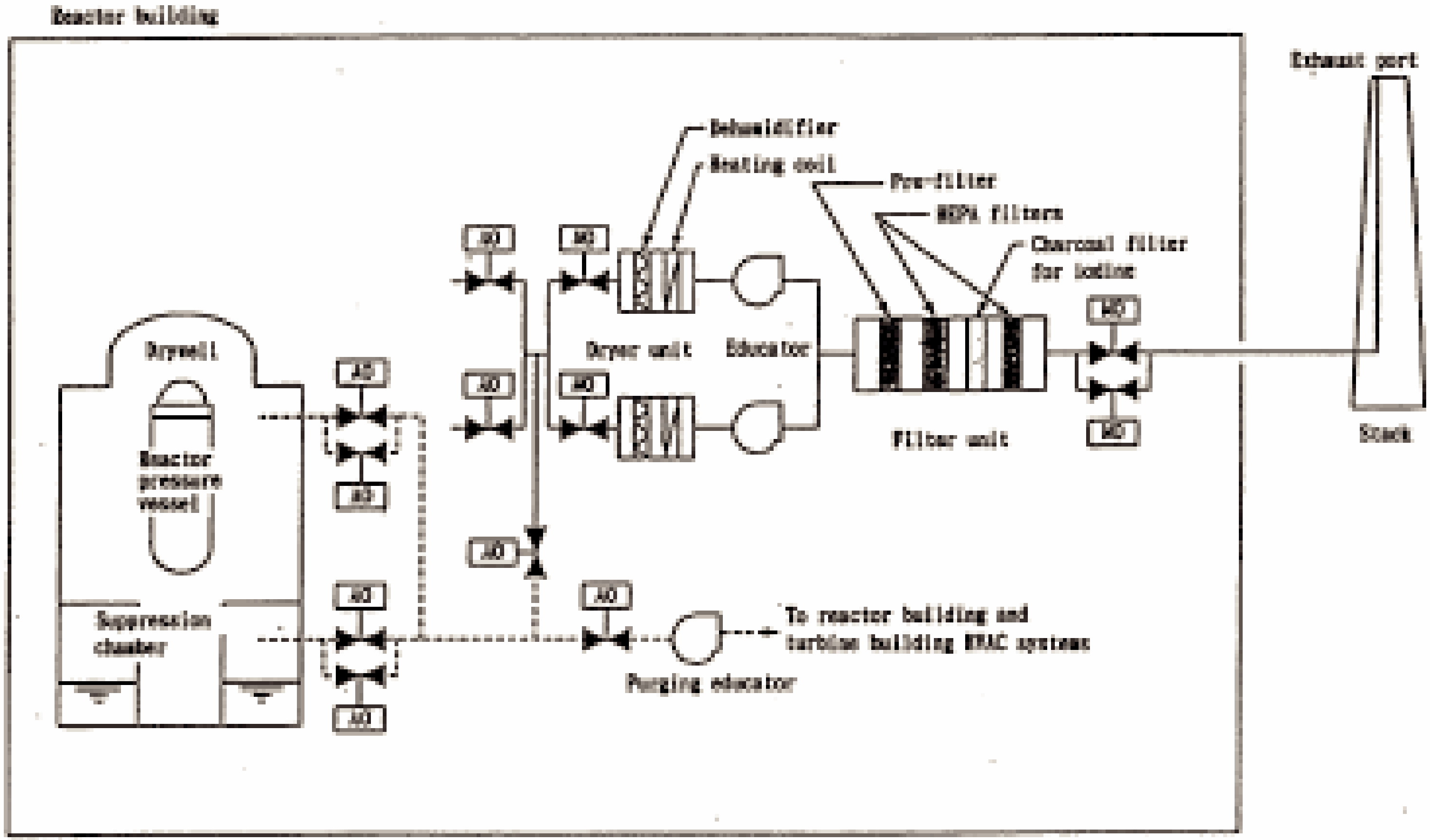


Fig.5.4 Schematic Flow Diagram of Emergency Gas Treatment System

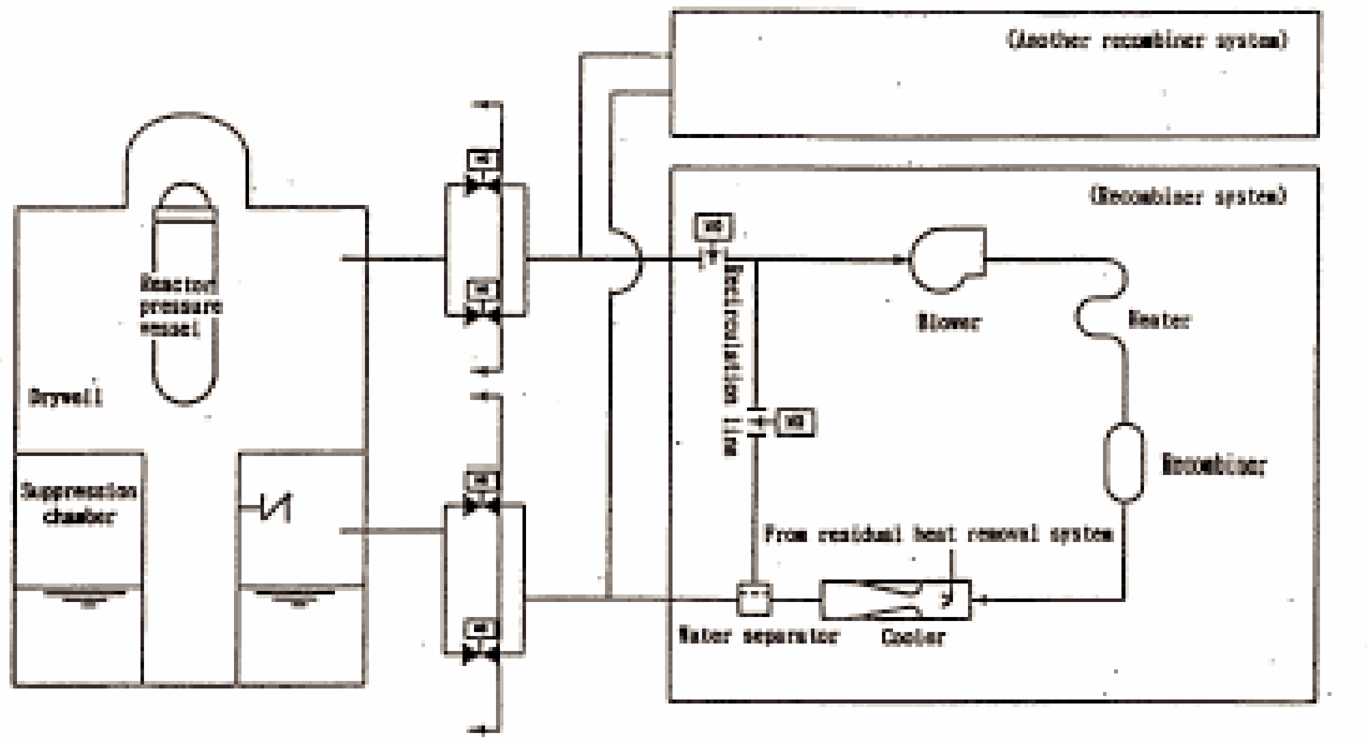


Fig.5.3 Schematic Flow Diagram of Flammability Control System

More details on Standard ABWR Technical Data（Source: GE HP）

Standard ABWR Technical Data

POWER	
Net electrical output	1356 MWe
Gross thermal power	3926 MWt
REACTOR CORE	
Active height	3.71 m
Active diameter	5.16 m
Number of fuel elements	872
Average power rating	196 W/cm
Average core power density`	50.6 W/litre
FUEL ASSEMBLIES	
Fuel material	UO_2, UO_2-Gd_2O_3
Average reload enrichment at equilibrium	3.2 %
Number of rods per assembly array	62
Fuel rod diameter	12.3 mm
Cladding material	Zircaloy 2
Cladding thickness	0.86 mm
Fuel discharge burnup (equilibrium)-reference case	32,000 MWd/t
CONTROL SYSTEM	
Number of control rods	205
Form of control rods	Cruciform
Neutron absorber	B_4C
Control rod drive motion	Electric-fine motion Hydraulic-scram
Burnable poison (Gd_2O_3)	
PRIMARY COOLANT SYSTEM	
Type	Internal recirculation pump system
Operating pressure	73.1 kg/cm^2
Feedwater inlet temperature	215.5 Degrees Celcius
Steam outlet temperature	287.4 C
Number of recirculation pumps	10
Recirculation mass flow (100% rated)	52,200 t/hr

REACTOR PRESSURE VESSEL

Internal height	21 m
Internal diameter	7.1 m
Wall thickness minimum	174 mm
Materials	Low alloy steel/stainless steel cladding

CONTAINMENT

Type	Reinforced concrete containment vessel (steel lined)
Design pressure	3.16 kg/cm^2
Height	36.1 m
Inside diameter maximum	29 m

TURBINE

Number	1
Maximum rating (@ 722 mm Hg)	1381 MWe
Speed	1500 rpm
Turbine inlet pressure	69.2 kg/cm^2
Turbine inlet temperature	283.7 C