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ADVANCED BOILING WATER REACTOR

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Abstract

In the Boiling Water Reactor (BWR) system, steam generated within the nuclear boiler is sent directly to the main turbine. This direct cycle steam delivery system enables the BWR to have a compact power generation building design. Another feature of the BWR is the inherent safety that results from the negative reactivity coefficient of the steam void in the core. Based on the significant construction and operation experience accumulated on the BWR throughout the world, the ABWR was developed to further improve the BWR characteristics and to achieve higher performance goals. The ABWR adopted "First of a Kind" type technologies to achieve the desired performance improvements. The Reactor Internal Pump (RIP), Fine Motion Control Rod Drive (FMCRD), Reinforced Concrete Containment Vessel (RCCV), three full divisions of Emergency Core Cooling System (ECCS), integrated digital Instrumentation and Control (I&C), and a high thermal efficiency main steam turbine system were developed and introduced into the ABWR.

1. INTRODUCTION

The first ABWR plants were constructed for the Tokyo Electric Power Company (TEPCO) Kashiwazaki-Kariwa Nuclear Power Station Units No.6 and 7 (K-6 and K-7) as a joint venture effort of GE, Hitachi, and Toshiba. The construction started in September 1991 and February 1992, with commercial operation starting in November 1996 and July 1997, respectively. Both units have now completed the first full fuel cycle of operation, and the desired performance improvements have been achieved by these first units. A view of the site area is shown in figure 1.

The following provides a general description of the ABWR and the construction experience of the units at K-6 and K-7.

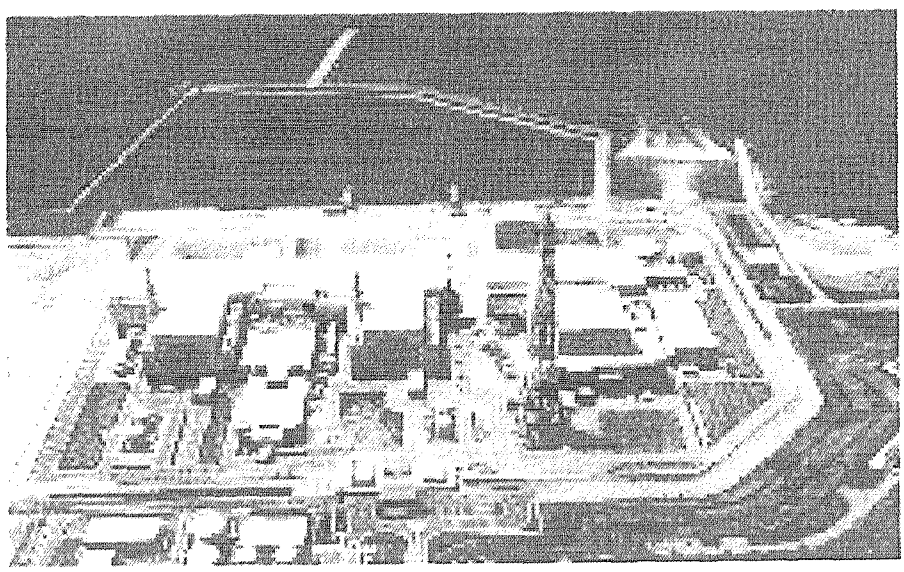

FIG. 1. View of TEPCO Kashiwazaki-Kariwa N.P.S. Unit 5, 6, 7 (from right to left).

2. GENERAL SPECIFICATION

The ABWR was designed to have the range of 1,350 - 1,380 MW electrical output with a nuclear boiler thermal output of 3,926 MW. The actual K-6 and K-7 gross electrical output is 1,356 MW based upon the local sea water temperature. The thermal output is generated by the nuclear core that consists of 872 fuel bundles.

The ABWR was designed as a twin unit plant. Each unit has its own Reactor Building and Turbine Building while the Control Building, Service Building, Radioactive Waste Building, and other utility facilities are shared by the two units. The Reactor Buildings contain the RCCV, nuclear boiler systems, and the ECCS systems. The Turbine-Generator, main condenser, and associated condensate and feedwater systems are located in the Turbine Building. In case of K-6 and K-7, the Circulating Water Pump and auxiliary sea water cooling systems are located in the heat exchanger area which is a part of the Turbine Building. The Main Control Panels for both K-6 and K-7 are located in the Main Control Room of the Control Building so that one operating crew can operate both units. The Radioactive treatment systems and the Condensate Storage Pool are located in the Radioactive Waste Building. The Service Building is used for the protective clothing changing area and check points. The two Turbine Buildings and the Radioactive Waste Building are connected together by a single crane rail to enable all these buildings to be available for the turbine generator lay-down space during the refueling outages.

The ABWR design objectives included a 40 year plant life, 87% or better plant availability, less than 0.36 man-Sv per year occupational radiation exposure, less than 10^{-7} /reactor year core damage frequency.

The ABWR general specification is shown in Table 1, and the overall system diagram is shown in Figure 2.

TABLE-1 ABWR Major Plant Specification

ITEM	ABWR	Conventional BWR
Plant Output		
Thermal Output	3,926 MW	3,293 MW
Electric Output	1,356 MW	1,100 MW
Vessel Dome Pressure	7.17 MPa	7.04 MPa
Main Steam Flow	7,640 t/h	6,410 t/h
Turbine	TC6F-52in	TC6F-41in
Reheat	Two-stage Reheat	No Reheat
Power Density	50.6 kW/l	50.0 kW/l
Fuel Assembly	872	764
Control Rods	180-B_4C+25-Hf	168-B_4C+17Hf
Control Rod Drive		
Normal Operation	Fine Motion Stepping Motor	Hydraulic Locking Piston
Scram Insertion	Hydraulic with Stepping Motor Backup	Hydraulic
Recirculation System	10 Internal Pumps	2 External Pumps with 20 Jet Pumps
Core Flow (100%Rated)	52,200 t/h	48,300 t/h
Emergency Core Cooling System	3 Divisions	3 Divisions
Division 1:	RCIC+LPFL	LPCS+LPCI
Division 2:	HPCF+LPFL	LPCI+LPCI
Division 3:	HPCF+LPFL	HPCS
Primary Containment		
Type	Pressure Suppression	Pressure Suppression
Configuration	Cylindrical Reinforced Concrete Containment Vessel	Conical Freestanding Steel Vessel
Control and Instrumentation	Microprocessor based digital circuit	Relay-analog circuit
Data Transmission	Intelligent multiplexed fiber optics	Hard-wired cable

RCIC: Reactor Core Isolation Cooling HPCF: High Pressure Core Flooder
LPCS: Low Pressure Core Spray LPCI: Low Pressure Coolant Injection
LPFL: Low Pressure Flooder

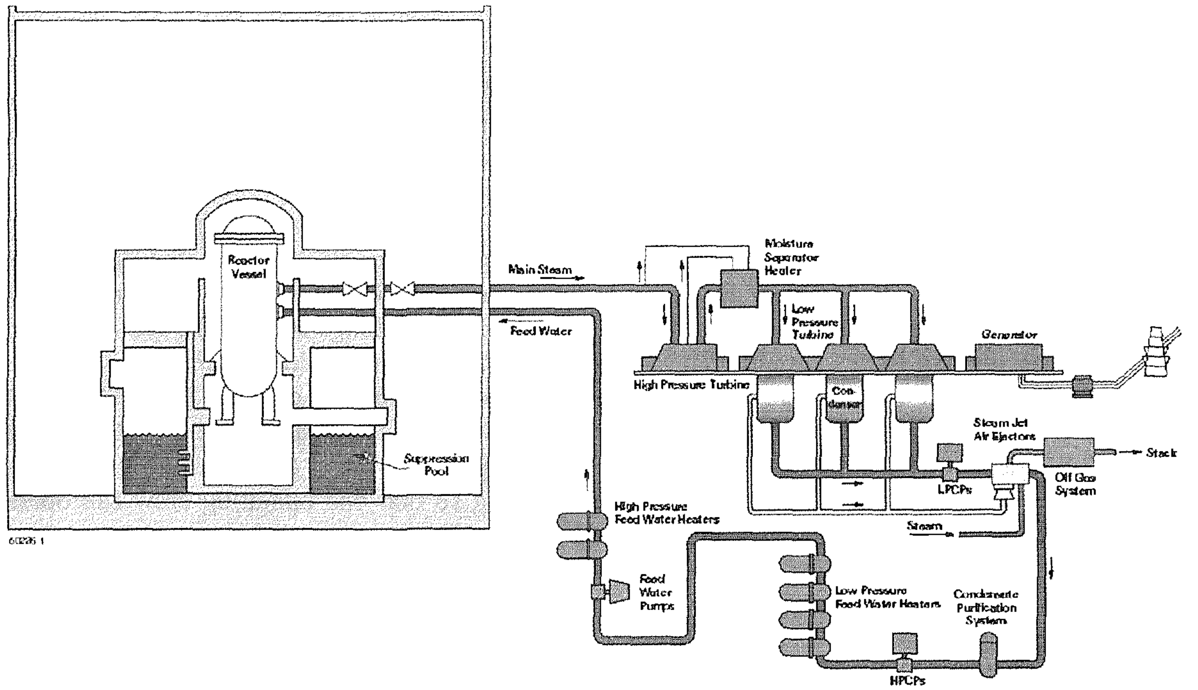

FIG. 2 ABWR overall system diagram

3. SAFETY FEATURES

3.1 Core Damage Frequency

In postulated Loss of Coolant Accident (LOCA) scenarios, core uncovery may result in core damage. Conventional BWRs utilize a defense-in-depth approach and their evaluated core damage frequency is less than 10^{-6}/ reactor year, which is already less than the 10^{-5}/ reactor year that is presented in the IAEA guidelines. However, for the ABWR, an even lower core damage frequency was desired.

A break in the piping which is connected to the Reactor Pressure Vessel (RPV) can result in a LOCA event. In case of a conventional BWR, the Primary Loop Recirculation (PLR) pump suction line is the largest pipe connected to the RPV and the postulated break of that piping is the limiting event for design basis LOCA analysis.

The ABWR adopted the Reactor Internal Pump (RIP) for the primary recirculation system instead of the external PLR design. With the elimination of the PLR lines, there are no longer any large bore pipes that connect to the RPV at an elevation below the top of the reactor core. Postulated LOCA scenarios were analyzed and the result was that a break in High Pressure Core Flooder (HPCF) line was determined to be the most severe event for the ABWR. Even in this case of an HPCF line break, it has been shown through analysis that the ECCS operation is sufficient to maintain the reactor coolant level above the top of the core. The Core Damage Frequency of the ABWR is calculated to be less than 10^{-7}/ reactor year.

With the elimination of PLR large bore piping, reactor pressure remains relatively high during the LOCA event. To assure early ECCS core injection, enforcement of the high pressure injection system was desired. The ECCS network in the ABWR is composed of three high pressure injection systems and three low pressure injection systems. However, the capacity of each system has been reduced from that in a conventional BWR as a result of the design basis LOCA flow rates being significantly reduced in the ABWR design.

3.2 Anticipated Transient Without Scram

The rapid Control Rod (CR) insertion into the core (scram) is initiated to terminate or mitigate a broad range of plant transient and postulated accident conditions. The Control Rod Drive (CRD) system has been designed to achieve that scram function with a high degree of assurance. Nevertheless, the Anticipated Transient Without Scram (ATWS), in which the scram function is postulated to fail to function properly, is considered in the ABWR design.

In a conventional BWR, hydraulic pressure is used for the normal insertion and withdrawal of the CR, and also for the scram function. Under normal scram conditions, high pressure water that is stored in an accumulator is discharged to the underside of the Control Rod Drive (CRD) causing it to rapidly insert (i.e., scram). The discharge of that high pressure water to each CRD is controlled by a Scram Valve. The Scram Valve is an air-operated valve that is actuated by the operation of a Scram Solenoid Valve (SSV). The Reactor Protection System (RPS) controls the operation of the SSV. In an ATWS event, the SSVs are postulated to have failed such that they do not operate to release the air pressure holding the Scram valves shut. As an ATWS mitigating measure, backup scram solenoid valves (of a diverse design) are installed on the Instrument Air line. Operation of the backup scram solenoid valves by the RPS causes the entire air supply line to be depressurized so that pressure controlling the individual Scram Valves is released and the high pressure water can be delivered to the CRDs as required for the scram operation.

The ABWR adopted the FMCRD which uses a stepping motor for the normal CR insertion and withdrawal function. But the hydraulic pressure is used for the scram function like a conventional CRD system. The back up scram valves, like those in a conventional BWR, are also applied. In addition to that back-up scram feature, in the ABWR, the stepping motor is activated automatically upon scram initiation. This motor driven CR run-in function provides an additional degree of ATWS mitigation.

3.3 Severe Accident Mitigation

Features which specifically address Severe Accident mitigation have been incorporated into the ABWR design. To provide an additional means of injecting water to the core under severe accident scenarios, facilities are provided to inject Make Up Water (MUW) system water to the RPV via connections in the Residual Heat Removal (RHR) system piping. To provide additional means for replenishing the water available to the ECCS special facilities are provided to connect the Fire Protection (FP) system to the MUW system and to connect the MUW system to the Suppression Pool which is inside the RCCV and is the primary suction source for the ECCS. Finally, to avoid overpressurization of the RCCV, a feature has been added to the vent system to bypass the Standby Gas Treatment System (SGTS) filter trains so that the RCCV can be vented directly through the main stack.

4. CORE AND FUEL

The latest Japanese BWR fuel bundle design consists of an 8 x 8 or 9 x 9 array of fuel pins surrounded by a fuel channel box. Inside each fuel pin, small pellets of sintered Uranium oxide are stacked. The fuel pins are supported by upper and lower tie plates at both ends and an intermediate positions by spacer assemblies to form a fuel bundle. The fuel channel box forms a flow channel for the reactor coolant. Throughout the development of fuel performance improvements, BWR fuel has maintained its basic physical dimensions so that application of new fuel designs to the existing fleet of operating BWRs is always possible.

In previous fuel designs, the uranium concentration was uniform alongside axial location of the fuel pin. Maintaining power as the fuel burns up is accomplished by the relatively frequent insertion and withdrawal of Control Rod throughout the core. However, the current advanced fuel designs have variable Uranium concentrations axially from the upper regions of the individual fuel pins to the lower regions. With this new type of fuel and the application of Control Cell core management methods, only the Control Rods in the Control Cell locations are used for power compensation. The ABWR has 25 Control Cells, and Control Rods with Hafnium blades are used in these Control Cell locations. Conventional Control Rods with Boron Carbide blades are used in the other Control Rod locations, which are used only for the purpose of plant shut down.

The BWR core is unique, in comparison to other Light Water Reactor (LWR) designs, in that the steam voids generated within the core perform an important role in controlling the core reactivity. The uranium fission process generates only fast neutrons and, in order to achieve a sustained chain reaction, these fast neutrons must be moderated to become thermal neutrons. In the BWR core, water acts as the moderator. When the steam void increases in the core, reactivity will decrease because of the decreased moderation function of the water. Increased RIP speed, which corresponds to increased core flow, will drive steam voids out from the core. In this condition, the reactivity will increase due to the increased water volume. Reactor power change is accomplished by capitalizing upon this inherent BWR negative power coefficient. An increase in recirculation flow temporarily reduces the volume of steam in the core by removing the steam voids at a faster rate. This increases the reactivity of the core, which causes the reactor power to increase. The increased steam generation rate increases the steam volume in the core with a consequent negative reactivity effect, and a new constant power level is established. When the recirculation flow is reduced, the power level is reduced in a similar manner.

5. POWER GENERATION

5.1 Nuclear Boiler

Moisture entrained in the steam, which is generated within the core, must be removed before the steam is delivered to the turbine. To remove that moisture, a Steam Separator and a Steam Dryer are located inside the RPV directly above the core. The steam exiting the core flows upward into the Steam Separator, in which the moisture is removed using the upward centrifugal movement of the steam. The Steam Dryer, which is located directly above the Steam Separator, further removes the moisture using the impingement vanes. The dry steam exits the RPV through a flow limiting orifice in the RPV nozzle of the Main Steam Line. The moisture accumulated within the Steam Separator and Steam Dryer is discharged to the annulus region between the Core Shroud and RPV, and returned to the core by the Reactor Internal Pump.

6. TURBINE GENERATOR

The ABWR has a 1,540MVA capacity Main Generator which is driven by three Low Pressure Turbines and one High Pressure Turbine in tandem compound configuration.

The steam from the nuclear boiler goes into the High Pressure Turbine through the Main Stop Valve and Control Valve. The High Pressure Turbine exhaust steam passes through the Moisture Separator ReHeater (MSH). The MSH removes the moisture from the steam, and the dry steam is heated by two stages of extraction steam. Within the MSH, the moisture is removed utilizing impingement vanes similar to the Steam Dryer. The first stage heating steam is extracted from the High Pressure Turbine and the second stage is extracted from the main steam piping upstream of the High Pressure Turbine. The reheated steam is discharged to the Low Pressure Turbine through the Combined Intermediate Valve.

The Low Pressure Turbine adopted in the ABWR is one of the largest of its type. The conventional 1,100 MW class BWR have Low Pressure Turbines that have last stage blades that are 41 or 43 inches long, depending on the grid frequency. To achieve improved thermal efficiency, the ABWR adopted Low Pressure Turbines that have last stage blades that are 52 inches long. The Low Pressure Turbine exhaust steam goes downward into the Main Condenser where the steam is cooled by the sea water.

Three Circulating Water pumps send the sea water to heat removal units within the plant. In the Main Condenser, the sea water goes through titanium tubes to condensate the turbine exhaust steam. A Ball Cleaning facility is utilized to maintain the cleanliness of these titanium tubes.

6.1 Condensate and Feedwater System

The Condensed water is returned to the RPV by first passing through three 50% capacity Low Pressure Condensate Pumps, with one pump normally in stand-by. The condensate then passes through the Steam Jet Air Ejectors (SJAE). The SJAEs are used to remove non-condensable gasses from the condensate water and it is driven by auxiliary steam or the house boiler steam. The de-aerated water passes through the Gland Steam Condenser, which condenses the Turbine gland steam, and then it is filtered and demineralized by the Condensate Filter (CF) and the Condensate Demineralizer (CD).

The CF and CD remove foreign particles and ions from the condensate water to maintain the water quality. In the BWR system, it is very important to maintain the water quality from the view point of the corrosion prevention and occupational exposure reduction since the condensate water passes directly through the core. Three hollow fiber filter CFs remove particles from the water. The pre-coat type CF that is commonly used in a conventional type BWR has the capacity to remove both particles and ions. However, the pre-coat material contributed significantly to the generation of radio-active waste and so, the non pre-coat type filter was developed in Japan. The hollow fiber filters have been used at several of the latest Japanese BWRs constructed and their operating experience has been excellent. The resin bed Condensate Demineralizer performs the final removal of particles and ions from the water.

The demineralized water is pumped by the High Pressure Condensate Pumps through the four stages of Low Pressure Feedwater Heaters, which are located in the upper portion of the Main Condenser. Each of the three High Pressure Condensate Pumps have 50% capacity, with one pump normally in stand by. Each stage of the Low Pressure Feedwater Heaters has three heaters which corresponds to the three condenser rooms. The water exiting the Low Pressure Feedwater Heaters is pumped by the Reactor Feedwater Pump through the High Pressure Feed Water Heaters. Two 50% capacity Turbine Driven Feedwater Pumps are used during normal plant operations while other, i.e., 25% capacity motor driven feedwater pumps, are used during plant start-up and transient conditions. Two-stage High Pressure Feedwater Heaters are used for the final step in the heating of the feedwater.

In the conventional Japanese BWR plants, all the feedwater heater drains are discharged back to the Main Condenser. For ABWR, feedwater heater drain pump systems were adopted. The High Pressure Feedwater Heater drains are discharged into High Pressure Drain Pump system. The accumulated water in the high pressure drain tank is pumped by the High Pressure Heater Drain Pump into the condensate water system upstream of the feedwater pumps. The Low Pressure Feedwater Heater drains are cascaded into the Low Pressure Drain Pump system. The accumulated water in the low pressure drain tank is pumped by the Low Pressure Heater Drain Pump into the condensate water system upstream of the CD. With the application of these heater drain pump systems, overall system thermal efficiency is improved and the condensate water system capacity is reduced.

6.2 Electrical System

The 1,540MVA capacity Main Generator is driven by the Turbine. The generator consists of a stator (stationary armature winding and core) and rotating field. The excitation system provides DC current to make a rotating field within the generator. The Automatic Voltage Regulator (AVR) regulates the output Voltage of the generator at 27 kV.

The generator output voltage is then converted by the Main Transformer to 500 KV ultra-high voltage for connection to the electrical power grid.

During power operation, the station load is supplied from the Main Generator via two House Transformers. Safety-related equipment is connected to the off-site power network via a Start-up Transformer, and to the Emergency Diesel Generators.

7. EMERGENCY SYSTEM

7.1 Emergency Core Cooling System (ECCS)

The BWR ECCS system has had many variations. The latest BWR-5 units have three divisions of ECCS configured as shown below. The BWR-5 includes a Reactor Core Isolation Cooling (RCIC) system, but this system is not classified as an ECCS. Each division has its own Diesel Generator for emergency power supply.

- Division 1: Low Pressure Core Spray (LPCS) and Low Pressure Core Injection (LPCI)
- Division 2: LPCI and LPCI
- Division 3: High Pressure Core Spray (HPCS)
- Automatic Depressurization System (ADS)

The ADS is one of the Safety Relief Valve (SRV) functions and its purpose is to rapidly depressurize the reactor pressure so that Low Pressure Core Injection (LPCI) system operation is possible. The LPCI system is also used as a Residual Heat Removal (RHR) system during the normal plant refueling outages.

The ABWR strengthened the high pressure core cooling capability and has three full divisions of ECCS which are configured as shown below. The RCIC has been upgraded to be an ECCS. Through development study, it was confirmed that core spray is not mandatory to assure uniform core cooling. Based upon those results, all high and low pressure injection system employ core-flooders, instead of core sprays.

- Division 1: RCIC and Low Pressure Flooder (LPFL)
- Division 2: High Pressure Core Flooder (HPCF) and LPFL
- Division 3: HPCF and LPFL
- ADS

Each division has its own Diesel Generator for emergency power supply.

The HPCF directly injects water inside of the Core Shroud, while the RCIC provides water to the RPV through the feed water lines. Both of the high pressure injection systems uses the Condensate Storage Pool as their primary water source and the Suppression Pool as their secondary water source.

The LPFL has four different function modes. The low pressure core injection mode is used for the ECCS purpose. Other major operation modes are (1) shut down cooling mode, which is used during the plant shut down; (2) containment spray mode, which is used to suppress steam released into the Drywell during postulated LOCA events, and; (3) suppression pool cooling mode, which is used to cool the Suppression Pool water.

7.2 Reactor Containment

The free-standing steel Primary Containment Vessel (PCV) has been used in previous BWR designs. The PCV is designed to hold any fluids that could be discharged from the primary coolant system during a postulated LOCA event. The BWR uses a pressure suppression type containment, and a large volume of water is maintained in the PCV as a Suppression Pool.

In a conventional BWR-5, the PCV is separated by a diaphragm into a Drywell volume and a Wetwell volume. The Wetwell contains the Suppression Pool. Located in the Drywell is the RPV; the main steam and feedwater piping and; related valves, such as Main Steam Isolation Valve (MSIV) and SRV.

In case of a LOCA event, the steam released would fill the Drywell and would be discharged to the Suppression Pool through a series of connecting vent pipes. The released steam is condensed in the Suppression Pool. With this pressure suppression function, the PCV pressure is kept relatively low even during a LOCA event.

A drawback associated with the use of this steel PCV is that Reactor Building construction could not proceed until the PCV leak test is complete. Also the configuration restriction arising from the steel structure presented plant and equipment arrangement restrictions. To eliminate such issues, the Reinforced Concrete Containment Vessel (RCCV) was adopted for the ABWR. Instead of a 38 mm thick steel structure, as in the case of the conventional PCV, in the RCCV design a 6.4 mm steel liner plate is used for securing leak-tightness and 2 meter of reinforced concrete behind the liner plate is used for pressure containment. The reinforced concrete of the RCCV is integrally interconnected with the concrete floors and walls of the Reactor Building. With this configuration, the building construction can proceed simultaneously with the RCCV construction and there is greater flexibility in the design of the Drywell arrangement.

The conventional BWR has two big Primary Loop Recirculation (PLR) pumps and external loop piping which connect the pumps to the RPV. In the ABWR design, these external pumps and associated piping are replaced by the Reactor Internal Pumps (RIP) which are directly attached to the RPV. These changes in the reactor recirculation system enabled the required volume of the primary containment to be reduced and, at the same time, the RCCV center of to be lowered.

The Flammable Gas Control (FCS) system consists of a blower, heater, recombiner, and cooler. The FCS system is installed to prevent an explosive reaction between the hydrogen and oxygen that might be generated during a postulated LOCA event. Through operation of the FCS, the flammable gases are re-combined and the water produced is returned to the Suppression Pool.

The Standby Gas Treatment System (SGTS) consists of two trains of equipment. Each train consists of a dehumidifier, a high efficiency particulate filter, Iodine charcoal filter, and exhaust fan. The SGTS is installed to maintain the Reactor Building at a negative relative pressure and to remove suspended radioactive materials in the Reactor Building.

8. INSTRUMENTATION AND CONTROLS

The overall control system configuration diagram for the ABWR is shown in Figure 3.

8.1 Reactor Control Systems

The Reactor Control Systems are composed of three major systems. The Reactor Power Control System controls the reactivity, the Reactor Pressure Control System controls the reactor pressure, and the Reactor Feedwater Control System controls the reactor water level.

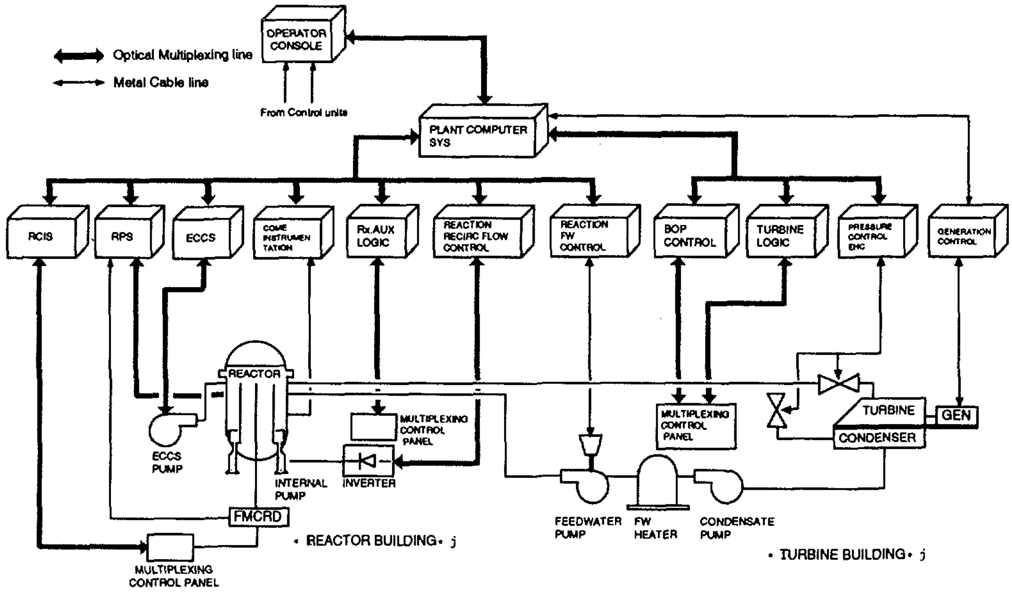

FIG. 3 ABWR Control System Configuration Diagram

8.1.1 Reactor Power Control System

The Reactor power is controlled by three elements: Control Rods position, recirculation flow, and the master controller signal from the main turbine.

The Rod Control and Information System (RC&IS) sends signals to the individual Fine Motion Control Rod Drive (FMCRD) mechanisms to withdraw or to insert. All Control Rods are inserted into the core when the plant is in shut down conditions. To start the plant operation, Control Rods withdrawal is necessary to increase the core reactivity by reducing the neutron absorption. With the application of the FMCRD and the associated RC&IS, multiple Control Rods can be simultaneously withdrawn (and inserted) and, as a result, less time is required to transition the plant from shutdown to normal operating conditions.

The core flow rate depends on the RIP rotating speed. The RIP is driven by an individual inverter controlled electric power source called an Adjustable Speed Drive (ASD). The Recirculation Flow Control (RFC) system sends a signal to control the RIP rotating speed based on the RFC master controller signal. Approximately 35% of the rated power level can be adjusted by the RFC control without any CR movement.

A signal from the speed-load governing mechanism to the master flow controller establishes the necessary reactor recirculation flow required to meet the system power requirements.

8.1.2 Reactor Pressure Control

The reactor pressure is controlled by the turbine Control Valve (CV) and Turbine Bypass Valve (TBV) opening control using the Electro Hydraulic Controller (EHC).

8.1.3 Reactor Feedwater Control System

The Feedwater Control System (FWCS) automatically controls the feedwater flow into the RPV to maintain the water at a predetermined level during all modes of plant operation. The FWCS utilizes signals from reactor vessel water level, steam flow, and feedwater flow.

8.2 Reactor Protection System

The Reactor Protection System (RPS) initiates CRD scram to insert all the Control Rods in the event of an abnormal plant condition.

The RPS is a four channel system which employs "two out of four" logic. This system is designed for maximum reliability to assure scram initiation is performed when necessary while minimizing the potential for unnecessary scrams.

Typical scram signals are listed below.

- High Pressure in the PCV.
- Low Water Level in the RPV.
- High Reactor Pressure.
- High Neutron Flux.
- Fast Closure of Turbine Control Valves.
- Closure of Turbine Main Stop Valve.
- Main Steam Isolation Valve Closure
- Steam Line High Radio-activity
- Manual Scram

8.3 Neutron Monitoring System

The reactor power is monitored by a coordinated set of neutron monitoring (sub)systems which come into service depending on the neutron flux level.

8.3.1 Source Range and Intermediate Range

At plant start up, when the neutron flux level is quite low, neutron flux is monitored by the Source Range Neutron Monitor (SRNM) using a neutron countrate method. When the core reaches the intermediate neutron level, the countrate method is automatically changed to the voltage variance method.

The ABWR has 12 SRNM monitors, and they are inserted into the middle of the core through inverted thimbles.

8.3.2 Local Power Range Monitor

During power operation, the neutron flux is monitored by the Local Power Range Monitor (LPRM). There are 52 LPRM assemblies located throughout the core. Each LPRM assembly has four fission chamber detectors, located at fixed positions vertically in the core, and a calibration guide tube for a Traversing In-core Probe (TIP).

8.3.3 Average Power Range Monitor

The Average Power Range Monitor (APRM) is a four channel system which monitors the bulk average power of the core. Within each channel of APRM, the neutron flux signals from 52 LPRM detectors are averaged and the bulk power of the core determined.

8.3.4 Traversing In-Core Probe

For the calibration of the LPRM, the Traversing In-core Probe (TIP) ion chamber is inserted into each calibration tube of the LPRM assembly. Traversing the ion chamber through the LPRM develops a vertical neutron flux profile upon which the LPRM detectors at that location can be calibrated. The TIP index mechanism selects the LPRM to which the TIP is inserted.

8.4 The State-of-the-Art I&C Technology

8.4.1 Integrated Digital Control Technology

Japanese BWRs have made steady progress on the application of digital technology. It started with application to the radioactive waste treatment system controls, and then was applied to other non-safety systems. In the ABWR, safety related systems also apply digital technology and, thus, achieving a totally digital plant for the first time. The objectives for digital system application were;

- higher reliability with less components, self-diagnosis features, and elimination of sensor drift, and
- better maintainability with modular replacement, and
- reduced construction cost through reduced number of control panels and use of fiber optic multiplexing data system to reduce the amount of cabling required.

8.4.2 ABWR Main Control Room Panels

The ABWR main control panel has two major features. One of them is the operator console that has Cathode Ray Tubes (CRT) and Flat Displays (FD). Control of the entire plant can be completed from this operator console. The other feature is the Large Display Panel (LDP). The LDP is a solid mimic with the status of all major plant systems and equipment shown in the center of the panel. Important plant level annunciators are displayed at the left side of the panel while system annunciators are displayed on the top of the panel. The 110 inch large screen display is located on the right side of the panel on which any information from the individual CRTs can be displayed.

Plant operation is conducted by touch operation on the CRT or FD screens. The operator can access the individual control screens through multi-layer screen operation.

Operator guidance and automatic operation functions have been widely applied throughout the ABWR control room design. The operator work load was analyzed during the ABWR development stage. Through these studies, activities of heavy work load (such as plant stabilizing control after a scram) were identified and such heavy work load activities were automated.

During the plant start up, CR withdrawal and associated operations are conducted automatically after operator initiation. There are a series of pre-determined hold points where the plant condition will be monitored. At these hold points, the automated operation will stop and wait for the operator confirmation and action before automated actions can continue.

9. PLANT CONSTRUCTION AND OPERATION

The first ABWR was adopted by Tokyo Electric Power Company's Kashiwazaki-Kariwa Nuclear Power Station Units Nos. 6 and 7. The construction was completed as a joint venture effort of GE, Hitachi, and Toshiba.

The construction was achieved through the use of the latest construction technology, such as the all-weather construction method and the large block module construction method. During the plant pre-operational and start-up testing, it was confirmed that all the pre-determined ABWR performance goals were achieved.

K-6 started commercial operation on November 7, 1996 and K-7 on July 2, 1997. The first fuel cycle of operation for both units have been completed and the operating performance was excellent.

10. CONCLUSION

The first ABWR plants completed construction and started commercial operation in Japan. The construction experience and the operating performance showed the expected performance goals are achieved. With this result, two more subsequent ABWR units are now under licensing process and several ABWR units are in the planning stage in Japan.

Beyond Japan, the U.S. N.R.C. has issued the Design Certificate for the ABWR, two ABWR units have started construction in Taiwan, China, and the major components for these units are being manufactured.

It is expected that the ABWR will be the major nuclear power plant design in the coming century.