Completion of ABWR Plant
—Kashiwazaki-Kariwa Nuclear Power Station Unit Nos. 6 and 7—

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ABSTRACT: Advanced boiling water reactor (ABWR) development began in 1978, and ABWRs were first adopted for use as the No. 6 and No. 7 units of the Kashiwazaki-Kariwa Nuclear Power Station (K-6 and K-7) of Tokyo Electric Power Co., Inc. The commencement of commercial operations by K-6 and K-7, on 7 November 1996 and 2 July 1997 respectively, marked the first actual ABWR operations in the world. The majority of equipment for both the K-6 turbine island and the K-7 nuclear island was supplied by Hitachi, Ltd. ABWR nuclear power plants (which could also be called Japanese BWRs) represent a threefold improvement over conventional BWRs, and have attained favored status under Japan's BWR Improvement and Standardization Program. Many ABWR components and facilities differ from those of conventional BWRs, and shop tests, pre-operation tests and start-up tests have shown that they meet required functionality and soundness standards. Expectations are high that even with continuous commercial operation, ABWRs will continue to demonstrate excellent performance characteristics.

INTRODUCTION

ADVANCED boiling water reactor (ABWR) development is the result of joint cooperative efforts of international BWR (boiling water reactor) suppliers, under the guidance of Tokyo Electric Power Company. The year 1978 saw the first efforts to improve and technologically advance BWRs, with the formation of an advanced engineering team (AET) for this purpose. The basic ABWR design was drawn up by BWR suppliers in Japan, the United States, Sweden and Italy, the countries participating in AET at that time. In the Phase 2 step that followed, BWR suppliers in the United States and Japan completed a final, detailed ABWR design based on evaluations of the initial design. ABWR development then began in Japan, based on the country's experience in constructing and operating conventional BWRs. Today, they have attained status as the best and most successful type of Japanese BWR.

Throughout the history of BWRs, BWR equipment has been simplified and improved in attempts to reduce equipment costs. These improvements were achieved by applying technologies developed by BWR suppliers in the United States. As BWRs continued to be operated in Japan and in other countries, various other points needing improvement were discovered.

One result of this was the establishment in Japan of a BWR Improvement and Standardization Program as a means of addressing these problems. The program aims to achieve reduced radiation exposure and radioactive waste, and enhanced availability. Activities aimed at achieving these aims were performed throughout the duration of the first and second Improved and Standardization Programs for conventional BWRs.

Work began on the development of ABWRs during the same period, with the same aims as those of the Improvement and Standardization Program. Other aims were added as well, i.e. to achieve very substantial improvements in plant safety, reliability, cost efficiency and performance.

After the basic ABWR design was finished, the third Improvement and Standardization Program for ABWRs was established. The target was for ABWRs to attain the status of the best and most successful type of Japanese-made BWRs. To achieve this, reliability tests of new equipment were continuously performed with the support from the Japanese Government, and through these tests the new equipment was determined to be reliable and suitable for use in actual plants.

The first ABWRs developed in this process were adopted for use in the No. 6 and No. 7 units of

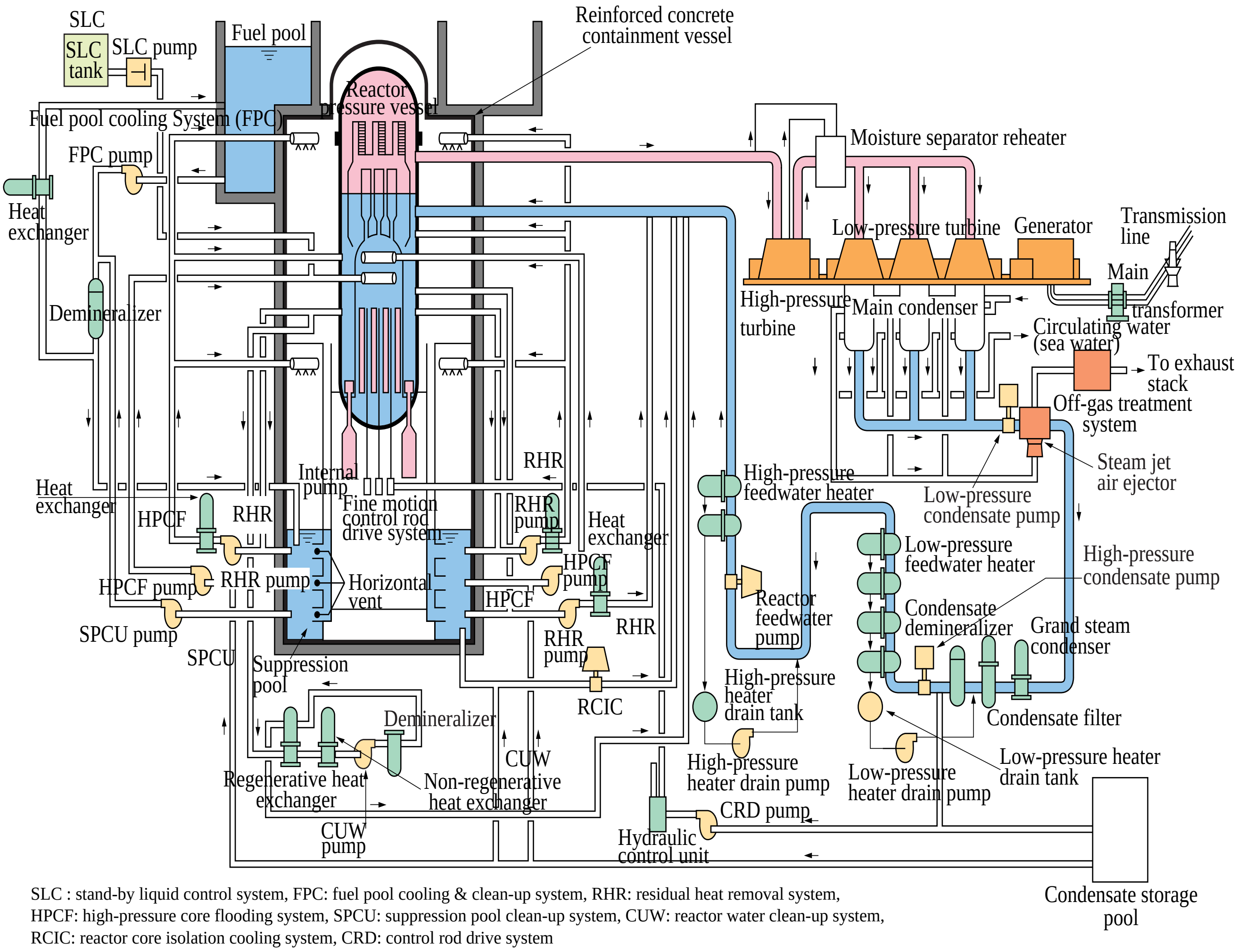

Fig. 1—ABWR System Outline.
The principal ABWR components are Reactor Internal Pump (RIP) system, Fine Motion Control Rod Drive (FMCRD) system, Reinforced Concrete Containment Vessel (RCCV), three divisions high pressure Emergency Core Cooling Systems (ECCSs), and high-efficiency turbine system.

Kashiwazaki-Kariwa Nuclear Power Station (K-6 and K-7) of Tokyo Electric Power Company. K-6 construction began in September 1991 and commercial operations commenced on 7 November 1996. K-7 construction began in March 1992 and commercial operations commenced on 2 July 1997. Fig. 1 shows ABWR system outline.

The beginning of commercial operations for the twin ABWR plants marked the end of a development period of approximately 19 years. The main equipment and technologies used in K-6 and K-7 are described in the following sections.

ABWR FEATURES

As stated in the previous section, the aims of ABWR development were (1) improved safety and reliability, (2) reduced radiation exposure, (3) reduced radioactive waste, (4) better operability, (5) enhanced cost efficiency, and (6) higher availability. To achieve these targets the following new systems are adopted in ABWRs.

(1) A reactor internal pump (RIP) system for the reactor coolant recirculation system (RRS)
(2) A fine motion control rod drive (CRD) system
(3) A reinforced concrete containment vessel (RCCV) with steel liner for the reactor containment vessel
(4) Three divisions high pressure emergency core cooling systems (ECCSs)
(5) A 52-inch last stage turbine, two-stage reheating cycle, and a heater drain pump-up system for the turbine system

The plant features resulting from the use of these systems are shown in Table 1.

TABLE 1. Plant Features Resulting from New Systems
Various new systems are adopted in ABWR plants to achieve various desired features. Circles in the columns indicate that the system contributes to the features listed in the rows.

Feature \ System	RIP	FMCRD	RCCV	Three divisions high pressure ECCS	Higher performance turbine system
High safety/ high reliability	○	○	○	○	—
Reduced radiation exposure	○	○	—	—	—
Less low radioactive waste production	—	—	—	—	○
Higher performance	○	○	—	—	—
Enhanced cost efficiency	○	—	○	—	○
Higher availability	—	○	—	—	—

MAJOR ABWR SYSTEMS

Table 2 lists the major ABWR specifications. The major ABWR systems are described in the following subsections.

RIP System

In the RIP system, pumps to recirculate the reactor coolant are directly mounted to the bottom of the ABWR reactor pressure vessel. This system is simple because no external recirculation piping or jet pump is required. The elimination of the external recirculation piping is particularly significant, as it provides a wide space inside the primary containment vessel and removes a potential radiation source. As a result, work efficiency is enhanced and radiation exposure in maintenance work is reduced.

Furthermore, since there is no large-diameter nozzle below the core region of the reactor pressure vessel, there is no possibility of coolant being exposed to fuel even in the event of a LOCA, and thus safety is enhanced.

Fine Motion CRD System

The fine motion CRD system uses two different power sources: an electric motor drive for normal operation and a conventional hydraulic drive for emergency insertion (scram). If a scram is required, it is actuated by the hydraulic control unit. If the hydraulic control unit fails, the control rod insertion

TABLE 2. ABWR Main Specifications
ABWR adopts the RIP system for the reactor coolant recirculation system (RRS). RIP has a simple configuration having no external recirculation piping or jet pump.

Item		ABWR
Output	Electrical	1,356 MWe
	Thermal	3,926 MWt
Core	Fuel	872
	Control rods	205
Reactor facilities	Recirculation system	Internal pump system
	Control-rod-drive system	Both hydraulic and electric-motor -drive systems
Primary containment vessel		Reinforced-concrete integrated with reactor building
Emergency core cooling system		Three divisions high-pressure systems
Turbine facilities	Heating cycle	2-stage reheating
	Turbine (last stage blade)	52 inch
	Moisture separator	Reheating type
	Heater drain	Drain pump-up system

TABLE 3. Main Specifications of K-7 RIP System and Related Equipment
The RIP system controls the reactor output in accordance with the inverter (ASD) frequency. Because the rotating parts are operated within the reactor vessel, they are required to have high reliability.

Equipment		Item	Specification
Internal pump	Pump	Quantity	10
		Rated flow, head	7,700 m^3/h, 40 m
		Max. speed	1,490 rpm
	Motor	Type	Submerged motor
		Output	830 kW
RIP power supply unit	Inverter	Quantity	10 (one per RIP)
		Type	GTO inverter
		Rated capacity	1,250 kVA
Motor generator set	Synchronous generator	Quantity	2 (one per 3 inverters)
		Rated capacity	5,100 kVA
	Driving motor	Quantity	2
		Rated output	3,800 kW

GTO: gate turn-off thyristor

will be actuated by the electric motor control unit. Thus a fully reliable emergency insertion system is ensured.

Furthermore, in normal operation control rods are moved in 18-mm steps, and each step raises core thermal power only slightly. Thus the control rods can be operated even at high power levels, further enhancing operability.

Finally, since simultaneous operation of the multiple control rods (gang mode) is possible, the plant start-up time is shortened and availability increases as a result.

Reinforced Concrete Containment Vessel (RCCV)

The use of reinforced concrete provides sufficient strength to withstand the high internal pressure postulated during accidental condition, and the use of a steel liner ensures the required air seal is maintained. Because the steel containment vessel in conventional BWRs must perform both functions, the steel must be very thick. In the ABWR RCCV, however, the steel mass is reduced since only a thin steel liner is required.

Three Divisions High Pressure ECCSs

Since there is no large-diameter nozzle below the core region of the reactor pressure vessel, the water level would decrease very slowly even in the event of a LOCA. Reactor pressure would also decrease slowly, however, and thus the injected water needs to be maintained under high pressure. Accordingly, all three sections of the emergency cooling system have a high pressure injection system. This ensures that in the event of a loss of coolant accident, the core flooding will be maintained and safety will be preserved.

The residual heat removal system is also divided into three divisions. With this type of system, according to the results of a probabilistic safety assessment, core melt frequency will be less than in conventional BWR, and thus safety is enhanced.

High Efficiency Turbine System

A high efficiency turbine system is adopted, featuring the use of a 52-inch long blade for the last stage of the turbine, a two-stage moisture separator-reheater, and a heater drain pump-up system connected to the condensate system. Thermal efficiency is enhanced through the use of this system.

Main Systems and Performance Tests

Reactor Internal Pump (RIP)

Structure and function

The RIPs are submerged, vertical type, single stage, mixed-flow pumps. The pump part consists of an impeller shaft and diffuser, and the submerged motor part consists of a stator, rotor, and bearings. The RIPs are encased by 10 motor casings attached to the bottom of the reactor pressure vessel. With its use of an adjustable speed drive (ASD), the RIPs can change the reactor core flow rate by changing the pump speed, thus performing the important function of controlling the reactor output. Also, since the rotating parts of RIPs are operated within the reactor vessel, the pumps are required to have the utmost reliability. The main specifications of the RIP system and related equipment are shown in Table 3.

Performance and function tests

After the fabrication of the RIP system, the manufacturer conducted shop tests to verify the pumps' hydraulic characteristics, vibration characteristics, coast down characteristics, and anti-reverse rotation device function. Combination tests with the motor generator set were also conducted to confirm that the performance and functions were as required.

After the shop tests, the RIPs were installed into the reactor pressure vessel at the site, and start-up tests were performed. Characteristic tests which could not be done at the manufacturer's shop, i.e. tests under various operational modes using multiple pumps, were

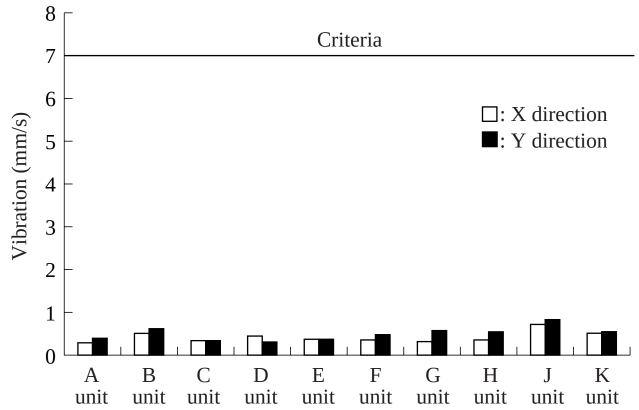

Fig. 2—K-7 RIP System Casing Vibration.
All pumps were confirmed to have a low and stable vibration level.

then performed, and it was confirmed that all requirements had been satisfied.

Fig. 2 shows the vibration characteristics obtained during the start-up tests. In each case, the result obtained satisfies the criterion, demonstrating that the RIP system has excellent characteristics. Furthermore, after the tests, the pumps were disassembled to check the rotating parts such as the bearings and impeller for detrimental flaws and wear. No such anomalies were found, thus confirming the structural integrity of the rotating parts.

These shop tests and at-site start-up tests demonstrated beyond any doubt that the RIP system satisfies all required specifications, and that RIPs are highly reliable pumps.

Fine Motion Control Rod Drive (FMCRD)

Structure and function

FMCRDs comprise the upper part (consisting of a hollow piston, ball screw and other parts) and the lower part (consisting of the shaft seal assembly, motor unit and other parts). A 1,356-MWe class ABWR is equipped with 205 FMCRDs, which are encased in housings attached to the bottom head of the reactor pressure vessel. The main FMCRD specifications are shown in Table 4.

TABLE 4. Main FMCRD Specifications

Item	Specification
Drive method	Normal : Electric motor driven Scram : Hydraulic pressure driven
Step size	Approx. 18 mm
Drive speed	30 mm/s
Scram time	Less than 1.44 s (at 60% insertion) Less than 2.80 s (at 100% insertion)
Number of drives in gang mode	Max. 26

Performance and function tests

For the prototype FMCRD and the hydraulic control unit (HCU), the following performance and function tests were conducted for the qualification.

- Endurance test for the designed 40-year life cycle
- Scramability test under simulated seismic conditions
- Various kinds of durability tests under severe conditions

These tests revealed that the all of the specified requirements were satisfied and that the prototype FMCRD had excellent performance and characteristics.

After the QC tests, FMCRDs were installed at the site and system performance tests were conducted under actual plant conditions during the preoperational and start-up test stages.

These tests under various plant operational conditions including the conditions not covered by the shop tests also revealed that FMCRD met the all specified requirements and that it was highly reliable equipment.

Typical scram time data are shown in Fig. 3.

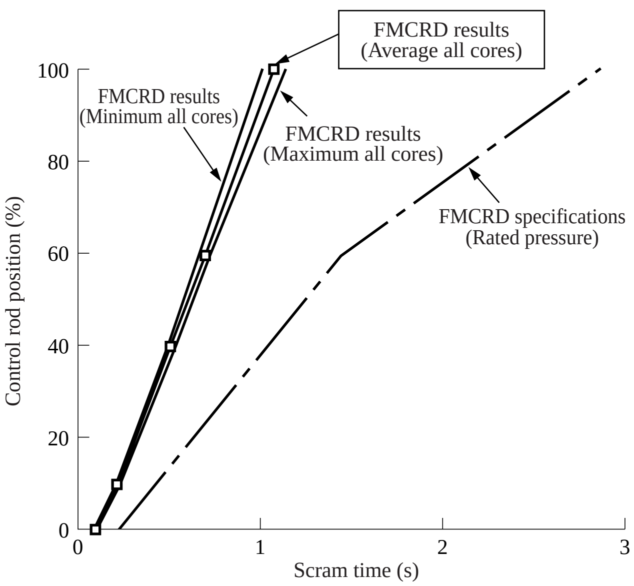

Fig. 3—Control-Rod Scram-Insertion Performance.
Scram insertion time for all 205 rods sufficiently satisfies requirements and demonstrates stable operation with little dispersion.

TABLE 5. K-6 Water Quality
All the K-6 water quality factors were found to be well under the required feedwater quality criteria.

Unit: ppb

Item	Required feedwater quality criterion	Feedwater	HP heater drain	LP heater drain
Fe	≤ 15	≤ 0.039	≤ 0.029	≤ 2.9
Ni		≤ 0.13	≤ 0.46	≤ 0.057
Co		≤ 0.0017	≤ 0.0019	≤ 0.012
DO_2	20~200	20~30	40~50	20~25

Reinforced Concrete Containment Vessel (RCCV)

Structure and function

The RCCV, a large vessel which contains important equipment (RPV, reactor pressure boundary piping , etc.), functions to suppress high pressure, prevent leakage of radioactive gases and liquids, and to provide a shield against radiation both during normal operation and in the event of an accident. As the RCCV is one of the most important components of a nuclear power plant, it is essential for it to have sufficient structural integrity and reliability.

Structural integrity test (SIT)

A SIT was performed to confirm that the RCCV had sufficient structural integrity. Displacement of several RCCV parts (e.g. top slab, pool girder) was measured, and it was verified that the concrete has sufficient structural integrity under pressure of 0.36 MPa, which corresponds to 1.125 times design pressure.

The SIT results confirmed that the maximum displacement of the parts measured is well below the stipulated criteria, and that the concrete part of the RCCV has sufficient integrity for the internal pressure loading. Over-pressure and leakage tests were also performed at the same time as the SIT, and the results confirmed that no abnormal deformation or leakage had occurred.

The results of all of these tests demonstrated that the RCCV has the required structural integrity, the required functionality to endure pressure and prevent leakage, and also high reliability.

Plant Thermal Efficiency

The basic ABWR turbine plant technology is that of the Kashiwazaki-Kariwa Nuclear Power Station Unit No. 4 of Tokyo Electric Power Company. In addition, other technologies and systems were applied to the plants to further improve plant thermal efficiency, e.g. the use of a 52-inch long blade for the last stage of the turbine, two-stage moisture separator reheaters, and a heater drain pump-up system.

Plant performance tests were conducted on K-6, and good results were obtained over a range of loads from partial to full. The electric power output of 1,356 MW and 34.5% plant thermal efficiency attained by K-6 are the highest levels yet measured in any Japanese BWR.

Heater Drain Pump-up System

A heater drain pump-up system is included in ABWR plants, the objectives being improved plant thermal efficiency and rationalized equipment specifications. The pump-up system comprises both high- and low-pressure systems. In the former, a high pressure feedwater heater drain is pumped into the condensate line in the suction side of the reactor feedwater pump. In the latter, a low pressure feedwater heater drain is pumped into the condensate line in the suction side of the condensate demineralizer.

Water quality

As stated above, in the heater drain pump-up system, a feedwater heater drain is pumped directly into the condensate system. Consequently, high water quality (in terms of metallic impurity level and DO2 concentration) must be maintained. The following measures were taken to ensure a high level of water quality.

(1) Improved materials were used in some of the turbine system equipment.

(2) The heater vent rate (the rate at which non-condensed gas is extracted from the feedwater heater and sent to the condenser) was optimized.

(3) A drain tank with a deaeration structure was used to ventilate non-condensed gas from the drain.

Actual data on K-6 water quality is shown in Table 5. All the K-6 water quality factors were found to be well under the stipulated feedwater quality criteria.

Stable drain pump operation

In the heater drain pump-up system, drain pump NPSH is given by the level difference between the drain tank and the drain pump.

In the high-pressure heater drain pump-up system, the heater drain temperature is extremely high. Consequently, it is essential to maintain high- pressure drain pump NPSH (HPDP NPSH) in cases where there is a sudden load decrease. Put another way, it is very difficult to maintain HPDP NPSH only through by the level difference between the drain tank and the drain

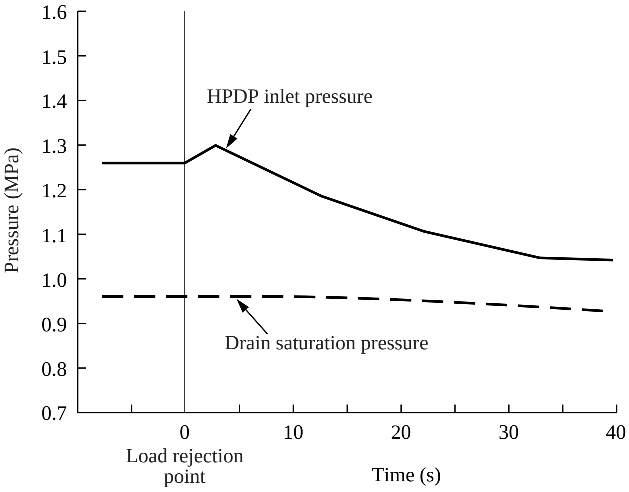

Fig. 4—Operation Data During 100% Load Rejection Test
HPDP inlet drain saturation pressure was lower than HPDP inlet pressure, confirming HPDP operation is stable.

pump, because changes in drain saturation pressure lag behind changes in the HPDP inlet pressure. As a countermeasure against this problem, Hitachi adopted a condensate injection system in which a portion of the low temperature condensate water is injected into the HPDP inlet pipe. With this system drain saturation pressure is lowered and HPDP operation is stabilized. A 100% load rejection test was conducted, and the results obtained are shown in Fig. 4. It was found that the saturation pressure in the HPDP inlet drain is lower than the HPDP inlet pressure, and thus stable HPDP operation is attained.

CONCLUSIONS

The main components and technologies adopted in Kashiwazaki-Kariwa Nuclear Power Station units No. 6 and No. 7 of Tokyo Electric Power Company are described. Judging from the results obtained in component tests and the from the good commercial operation of Kashiwazaki-Kariwa Nuclear Power Station units No. 6 and No. 7, we can confirm that ABWRs are excellent plants from a standpoint of quality and performance.

With the results we have obtained, our recommendation to any electric power company planning to construct a BWR is to construct an ABWR. In future work, we will attempt to use the experience we have gained in the design, manufacturing, construction and operation of Kashiwazaki-Kariwa Nuclear Power Station units No. 6 and No. 7 to plan and construct even better ABWR plants.

ACKNOWLEDGMENTS

The authors wish to express their thanks to Tokyo Electric Power Company and all electric power companies possessing BWRs for their help and cooperation during the 20 years it took us to successfully construct ABWR plants. Thanks are also due to the Ministry of International Trade & Industry and the Nuclear Power Engineering Corporation for their support in establishing the BWR Improvement and Standardization Program and conducting reliability tests of the new system.

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