CHAPTER

58

NEW GENERATION OF BWRS

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58.1 INTRODUCTION

Nuclear energy plays a major role in meeting the world's energy needs. At the end of 2005, there were 443 nuclear power plants operating in 32 countries, with 25 more units under construction. These plants account for 17% of world's electricity. In a nuclear power plant, steam to turn the turbine and in turn the generator to produce electricity is produced through a controlled nuclear fission reaction. The steam producing part of a nuclear power plant, including the supporting systems, is called the nuclear steam supply system (NSSS). Beyond the NSSS, the remaining part of a nuclear power plant, the turbine/generator and remainder of the steam cycle, is called the balance of plant (BOP). In a boiling water reactor (BWR) NSSS, the steam is directly produced in the reactor pressure vessel (RPV). In the pressurized water reactors (PWRs), the steam is produced in steam generators that are connected to RPV. Both the BWRs and PWRs are classed as light water reactors (LWRs) since they use light water (as opposed to heavy water, D_2O) as coolant and moderator. In the United States and many countries of the world, the RPV and the most of the NSSS components are designed, fabricated, tested, and inspected during service using the rules of ASME Boiler & Pressure Vessel Code. Specifically, Section II (for material selection and allowable stresses), Section III (for design and analysis), Section IX (for welding), and Section XI (for in-service inspection) of the Code are used. The overall objective of this chapter is to provide a description of the evolution of the BWR product line, including the current offerings (the Advanced Boiling Water Reactor, ABWR, and the Economic Simplified Boiling Water Reactor, ESBWR), along with the discussion on the role of ASME Code in the material selection, fabrication, design, and in-service inspection (ISI) of the BWR NSSS system.

The first section provides a general background of the development of the BWR product lines. This includes the description of the reactor and reactor system design, safety system design, and the containment design [1, 2]. The next section deals with the key features of the ESBWR, including the natural circulation design, operating domain, and passive safety features [3]. The ASME Code aspects are covered next, including the ASME Code versions, treatment of environmental fatigue issues, material selection, and others. Future directions in terms of fabrications, modularization, and others are discussed next. The last section provides a summary.

58.2 EVOLUTION OF BWR PRODUCT LINE FROM BWR/1 THROUGH ESBWR

The BWR nuclear plant, like the PWR, has its origins in the technology developed in the 1950s for the U.S. Navy's nuclear submarine program. The first commercial BWR nuclear plant to be built was the 5-MWe Vallecitos plant (1957) located near San Jose, California. The Vallecitos plant confirmed the ability of the BWR concept to successfully and safely produce electricity for a grid. The first large-scale BWR, Dresden 1 (1960), then followed. The BWR design has subsequently undergone a series of evolutionary changes with one purpose in mind – simplify.

The BWR design has been simplified in two key areas – the reactor systems and the containment design. Table 58.1 chronicles the development of the BWR.

58.2.1 General Progression of BWR Designs

Figure 58.1 illustrates the evolution of the reactor system design. Interestingly enough, Dresden 1 was not a true BWR. The design was based upon dual steam cycle, not the direct steam cycle that characterizes BWRs. Steam was generated in the reactor but then flowed to an elevated steam drum and a secondary steam generator before making its way to the turbine. The first step down the path of simplicity that ultimately led to the ESBWR was the elimination of the external steam drum by introducing two technical innovations – the internal steam separator and the steam dryer (KRB, 1962). The first large direct cycle BWRs (Oyster Creek and Nine Mile Point Unit 1) appeared in the mid-1960s and were characterized by the elimination of the steam generators and the use of five external recirculation loops to provide forced circulation flow through the core. Later, reactor systems were further simplified by the introduction of jet pumps inside the vessel for driving the core flow. The use of internal jet pumps allowed two external recirculation loops to drive the core flow, down from the five loops used in the large BWR/2 plants, thus reducing the associated piping, valves, pumps, and large vessel nozzles. This change first appeared in the Dresden 2 BWR/3 plant.

The use of reactor internal pumps in the ABWR design took this process of simplification another step further. By using internal pumps attached directly to the vessel itself, the jet pumps and the external recirculation systems, with all the associated large pumps, valves, piping, and snubbers, have been eliminated altogether. The

TABLE 58.1 EVOLUTION OF THE GE BWR

Product Line	First Commercial Operation date	Representative Plant/ Characteristics
BWR/1	1960	Dresden 1 Initial commercial-size BWR Dual cycle
BWR/2	1969	Oyster Creek Plants purchased solely on economics Large direct cycle Forced circulation Variable speed pumps for recirculation flow control
BWR/3	1971	Dresden 2 Internal jet pump application Improved ECCS: spray and flood capability
BWR/4	1972	Vermont Yankee Increased power density (20%)
BWR/5	1977	Tokai 2 Improved ECCS Valve flow control
BWR/6	1978	Cofrentes Compact control room Solid-state nuclear system protection system
ABWR	1996	Kashiwazaki-Kariwa 6 Reactor internal pumps Fine-motion control rod drives Advanced control room, digital and fiber optic technology Improved ECCS: high/low pressure flooders
ESBWR	Under review	Natural circulation Passive ECCS

development of the ABWR took place during the 1980s under the sponsorship of Tokyo Electric Power Company (TEPCO). In 1988, TEPCO announced that the next Kashiwazaki-Kariwa units (K-6 and 7) to be constructed would be ABWRs. The licensing activities for the K-6 and 7 were conducted with the Japanese regulatory agency, Ministry of International Trade and Industry (MITI), in parallel with the review of the ABWR in the United States by the Nuclear Regulatory Commission (NRC). K-6 entered commercial operation in 1996 and K-7 in 1997. Two more ABWRs entered commercial operation in Japan – Hamaoka-5 in 2005 and Shika-2 in 2006. In addition to these four units in Japan, two more ABWRs are being constructed for the Taiwan Power Company (TPC) at TPC's Lungmen site.

In the United States, the ABWR First-of-a-Kind Engineering (FOAKE) program was completed in September 1996. FOAKE represented a major step toward one of the U.S. industry's goals - to have a certified design that is 90% engineered prior to the start of construction. The ABWR Design Certification was signed into law on May 2, 1997 by the then Chair of the NRC, Shirley M. Jackson. In September 2007, NRG Energy submitted to the NRC a combined construction permit – operating license application (COLA) to build two ABWR units at the South Texas Project site in Texas.

The ESBWR and its smaller predecessor, the Simplified Boiling Water Reactor (SBWR), took the process of simplification to its logical conclusion with the use of a taller vessel and a shorter core to achieve natural recirculation without the use of any pumps. Following the Three Mile Island accident in 1979, there was significant interest in developing a reactor with passive safety features and less dependence on operator actions. Utilities also took this opportunity to request a reactor that was simpler to operate, had fewer components, and no dependence on diesel generators for safety actions. GE began an internal study of a new BWR concept

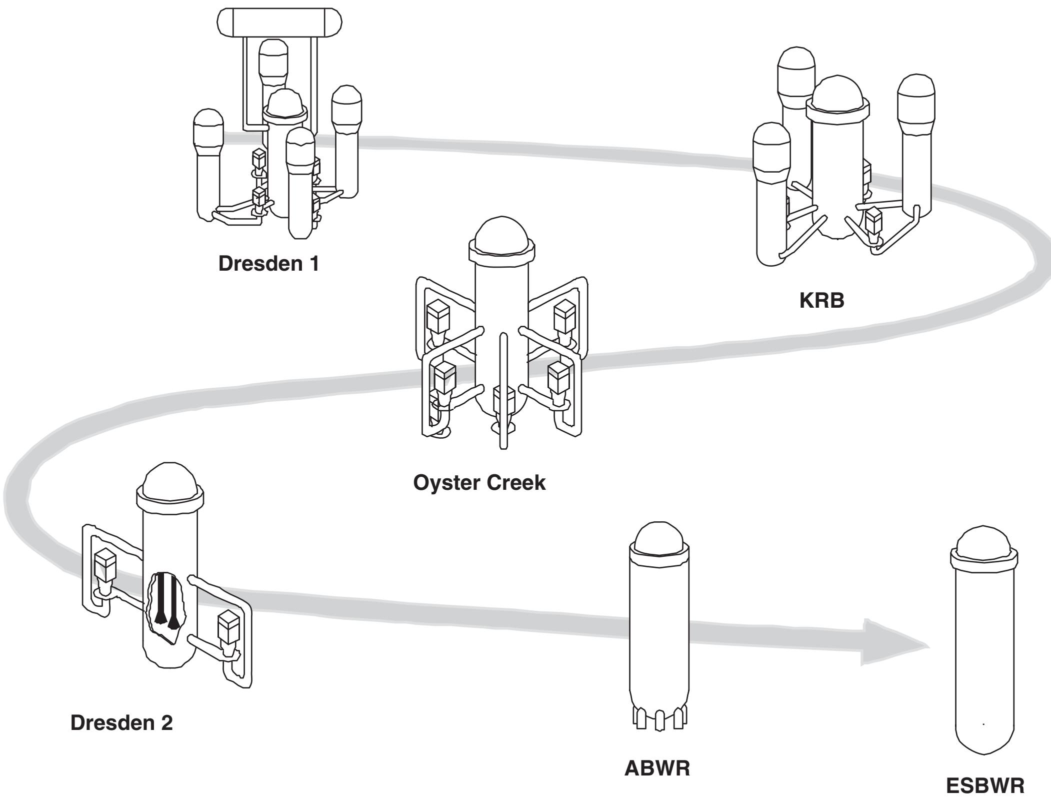

FIG. 58.1 EVOLUTION OF THE BWR REACTOR SYSTEM DESIGN [2]

based on these principles and the SBWR was developed in the early 1980s. This concept attracted development support from the U.S. Department of Energy (DOE), EPRI, and a number of U.S. utilities. Key new features, such as the gravity-driven cooling system (GDCS), depressurization valves (DPV), and leak-tight wet well/dry well vacuum breakers were tested. As interest grew, an international team was formed to complete the design, and additional separate effects, component and integrated system tests, particularly of the innovative new feature, the passive containment cooling system (PCCS), were run in Europe and Japan. A Design Certification Program was begun in the late 1980s with the objective of obtaining a standardized license, similar to that obtained for ABWR. However, as more of the design details became known, it became clear that at 670 Mwe, the SBWR was too small to generate the right economics for a newly build project. The certification program was stopped, but efforts continued to make the SBWR attractive for power generation. With European utility support, the SBWR was uprated gradually to its current power level of approximately 1550 Mwe. This was made possible by staying within the RPV size limit established by the ABWR and by taking advantage of the modular approach to passive safety afforded by isolation condensers (ICs) and PCCS.

The Design Certification application for the ESBWR was submitted to the U.S. NRC in August 2005 and was formally accepted for docketing in 3 months. The NRC's new plant review and licensing process has been improved, including allowance for parallel review of both the Design Certification and the COLA, with a focus on standardization and reducing and eliminating re-reviews of the same open items. GE is working with a number of customers who have selected the ESBWR technology and is participating in the U.S. Department of Energy's Nuclear Power 2010 program, which was established by the DOE to act as a catalyst for the newly built nuclear energy in the United States.

58.2.2 Reactor System Design

The first LWRs were pressurized to prevent the water from boiling in the core or turning to steam. The conventional wisdom was that steam bubbles would somehow affect the behavior of the neutrons and cause the reactor to behave erratically and possibly overheat. Samuel Untermyer, a scientist at Argonne National Laboratory, postulated that if water bubbled or steamed in an overheating reactor core, the chain reaction would slow down [4]. This concept of a negative void reactivity coefficient proved to be fundamental to the successful design and operation of the BWRs. Untermyer initiated the Boiling Water Reactor Experiment (BORAX) series of test reactors at the National Reactor Testing Station in Idaho. In 1953, the first test reactor in the series, BORAX-1, demonstrated two theories: (1) boiling water in a reactor was a stable condition, and (2) the void coefficient can control the power increases. Argonne modified the BORAX reactor design several times while conducting experiments to develop an understanding of the parameters necessary for operating a BWR safely.

Argonne's ultimate goal was to evolve a reactor useful for electrical generation. The early experiments with the BORAX reactors demonstrated the inherent safety and stability of the BWR. Using a turbine generator scrounged from an old sawmill, the BORAX-III reactor lit the town of Arco, Idaho, on July 17, 1955 with nuclear power for the first time in history. The first BWR power plant, EBWR (Experimental BWR), built by

Argonne Laboratory, was designed for 20 MWe. The plant ran from 1956 to 1967, gradually increasing its power level and reliability from the point where it supplied electricity to the entire Argonne Lab.

The main characteristic of the BWR is the fact that bulk boiling takes place in the reactor core. Three configurations have been used for the power cycle design of the BWR plant. In the direct cycle BWR, the simplest design, the steam is passed directly from the reactor vessel to the turbine. In the indirect cycle design, the steam–water mixture passes through a steam generator; the steam produced on the secondary side of the steam generator is then passed to the turbine. The indirect cycle design has the advantage of isolating the radioactive contaminants within the primary system; however, this advantage is offset by the increased cost and maintenance associated with the steam generators. The dual cycle design is a combination of both the direct and indirect cycles, with high-pressure steam being passed directly to the high-pressure end of the turbine and lower pressure steam from the steam generator entering the turbine at an intermediate stage. The dual cycle combined reliability, stability, a fairly high power density, and a well-controlled reactor response to changes in load demand. The BWR designs can be further classified by the means used to drive the coolant through the core. In a forced circulation design, the core flow is driven by pumps external or internal to the reactor vessel. In a natural circulation design, the difference in density between the single-phase fluid in the downcomer region and the two-phase mixture in the core provides the driving force [5].

GE selected the BWR as the most promising nuclear power concept because of its inherent advantages in control and design simplicity. GE established an atomic power equipment business in 1955 to offer the BWR commercially. The first GE BWR, the Vallecitos BWR, was built in 1957. This 1000-psi reactor powered a 5-MWe generator and provided power to the Pacific Gas and Electric Company grid through 1963. The VBWR was constructed so that it could be operated in either a direct cycle mode or a dual cycle mode using either forced or natural circulation.

The advantages of the BWR compared to the PWR are as follows:

- The reactor vessel and associated components operate at a substantially lower pressure (about 1000 psig) compared to a PWR (about 2200 psig).
- There are no steam generators and no pressurizer vessel.
- Lower risk (probability) of a rupture causing loss of coolant compared to a PWR, and lower risk of a severe accident should such a rupture occur. This is due to fewer pipes, fewer large diameter pipes, fewer welds, and no steam generator tubes.
- The distance between the core and the vessel wall is greater; therefore, the pressure vessel is subject to significantly less irradiation compared to a PWR and so does not become as brittle with age.
- The fuel operates at a lower temperature.
- The single reactor vessel makes emergency conditions simpler to diagnose and emergency actions simpler to execute.
- A BWR may be designed to operate using only natural circulation so that recirculation pumps are eliminated entirely.

The disadvantages of a BWR are as follows:

- The core nuclear and thermal hydraulic calculations are more complex when considering two-phase flow.
- The RPV is much larger than that of a PWR of similar power. However, the overall cost is reduced because a modern BWR has no main steam generators or pressurizers.
- The primary coolant passes through the turbine, contaminating it with short-lived activation products. Therefore, radiation exposure must be managed in the vicinity of the steam piping and turbine.

The first commercial BWRs were small demonstration plants for evaluating various approaches to reactor and containment design and for demonstrating key system design features. These plants are referred to as the BWR/1 product line, even though they were custom designs. Dresden 1 was the first commercial BWR. Dresden 1 was based upon dual cycle design and used forced circulation. The plant used a spherical dry containment. Steam was generated in the reactor but then flowed to an elevated steam drum and a secondary steam generator before making its way to the turbine. The dual steam cycle system was employed to assure overall plant stability between the reactor and turbine generator during normal operation and operational transients. Dresden 1 was the model for several other dual cycle plants (Garigliano, KRB, Tarapur). Humboldt Bay and Big Rock Point were the first commercial direct cycle plants.

Table 58.2 provides a comparison of the key features of the commercial BWRs. The first standardized BWR design was the BWR/2 (Oyster Creek and Nine Mile Point 1). The BWR/2 direct cycle design used the internal steam separator and steam dryer demonstrated at KRB. This simplified the reactor design by eliminating the external steam drum and steam generator pressure vessels. The steam separator and dryer assemblies are removed to allow access to the core during refueling. The control blades enter the core from the bottom of the vessel. Control rod entry from the bottom of the core provides the best axial flux shaping and resultant fuel economy for the BWR. The bottom entry drives do not interfere with refueling operations. The piston drive and high-pressure scram accumulators provide high scram forces and ensures rapid rod insertion.

The BWR/2 design also standardized the use of forced recirculation core flow. Oyster Creek and Nine Mile Point 1 passed the entire core flow through five external recirculation loops with pumps. Forced recirculation core flow provides a second means of controlling core power in addition to the control rods. Forced recirculation allows the core power to be controlled by changing recirculation flow by exploiting the negative void reactivity characteristics of the BWR core design. An increase in the core flow increases the liquid in the core, which increases the neutron moderation and power generation. The core stabilizes at a higher power level. The ability to change power gradually by using core flow also reduces the thermal mechanical duty on the fuel. The control rod drives (CRDs) move the control blades in finite increments, which may result in a significant local power increase near the tip of the control blade. Varying the core flow also varies the boiling boundary in the core. This allows spectral shift core management schemes to be employed. Early in the cycle, the core flow is kept at low, lowering the axial flux shape and boiling boundary in the core, which increases the voiding in the top of the core. This voiding hardens the neutron spectrum in the upper part of the core resulting in the breeding of plutonium. At the end of the cycle when the reactivity in the lower part of the core is exhausted, the core flow is increased, shifting the axial flux shape upward and burning the plutonium generated in the upper part of the core early in the cycle. Thus, the ability to control core power

TABLE 58.2 COMPARISON OF KEY FEATURES OF GE BWRs

Parameter	BWR/2	BWR/4	BWR/6	ABWR	ESBWR
Power (MWt/MWe)	1930/670	3293/1098	3900/1360	3926/1350	4500/1550
Vessel height/diameter (m)	19.5/5.4	21.9/6.4	21.8/6.4	21.1/7.1	27.7/7.1
Fuel bundles, number	560	764	800	872	1132
Active fuel height (m)	3.7	3.7	3.7	3.7	3.0
Power density (kW/L)	40.5	50	54.2	51	54
Recirculation pumps	5 (external)	2 (external)	2 (external)	10 (internal)	0
Number/type of CRDs	137/LP	185/LP	193/LP	205/FM	269/FM
Safety system pumps	12	9	9	18	0
Safety diesel generators	2	2	3	3	0
Alternate shutdown	2 SLC pumps	2 SLC pumps	2 SLC pumps	2 SLC pumps	2 SLC accumulators
Control and instrumentation	Analog single channel	Analog single channel	Analog single channel	Digital multiple channel	Digital multiple channel
Core damage (freq/yr)	10^{-5}	10^{-5}	10^{-6}	2×10^{-7}	3×10^{-8}
Safety bldg vol (m^3/Mwe)	110	120	170	180	130

using recirculation flow ultimately results in better fuel cycle economics for the plant.

The next significant simplification introduced in the reactor system was the internal jet pump in the BWR/3 design. These pumps sufficiently boosted recirculation flow so that only two external recirculation loops were needed. This change was first incorporated in the Dresden 2 plant. The use of internal jet pumps resulted in the elimination of several of the recirculation lines and pumps used previously in the BWR/2 reactors. Savings in investment for recirculation lines and pumps were partially offset by the higher power requirements due to the lower efficiency of the jet pump. More notably, the use of jet pumps gives some safety advantages in that the number and size of major nozzle penetrations on the vessel is reduced and the reactor internal arrangement allows the capability of reflooding the vessel and maintaining coolant level in the core even if there is a complete severance of the recirculation line. Increased natural circulation capability was also gained with the jet pump system.

The basic jet pump BWR design was carried through most of the plants comprising today's BWR operating fleet. The BWR/4 design (Browns Ferry and Peach Bottom) introduced improved ECCS and higher core power densities. The BWR/5 design (Tokai 2) introduced constant speed recirculation pumps and flow control valves used to control the reactor core flow and improved ECCS. The change from variable speed recirculation pumps to constant speed pumps with valve control allowed the plants to follow more rapid load variations and reduced the capital cost of the overall control system. The BWR/6 design (Cofrentes) incorporated a five-hole nozzle design in the jet pump that increased the pumping efficiency and allowed the use of smaller jet pumps. With smaller jet pumps, the size of the core could be increased, increasing the reactor output for the same size vessel.

The ABWR design (Kashiwazaki-Kariwa 6) incorporated reactor internal pumps to drive the core flow. By using pumps attached directly to the vessel itself, the jet pumps and the external recirculation systems, with all their pumps, valves, piping, and snubbers, have been eliminated altogether. With the elimination of the large external recirculation piping, there were no large pipes attached to the vessel below the top of the core. This allowed the ABWR to be designed to keep the core submerged and cooled during the break of any pipe forming the reactor coolant pressure boundary. Fine motion CRDs permit small power changes, improved start-up times, and improved power maneuvering.

The ESBWR design uses natural circulation to provide the driving force for the core flow, eliminating the recirculation pumps altogether. Building on the basic ABWR vessel design, the height of the vessel has been increased to provide the additional driving head necessary to achieve the core flows and power levels necessary for a modern reactor. Variable feedwater temperature control has been added to provide a second means of reactivity control in addition to the fine motion CRDs. Figure 58.2 shows the direct cycle power conversion system used in most BWRs.

58.2.3 Safety System Design

The loss-of-coolant accident (LOCA) has dominated the design of the safety systems for BWRs. A LOCA is defined as a breach in the primary coolant system pressure boundary. The design basis accident (DBA) was specified to be the instantaneous double-ended guillotine break of the largest pipe attached to the RPV. However, the safety systems must also be designed to mitigate the consequences of the full spectrum of potential break sizes and locations anywhere in the primary coolant piping, as well as equipment failures such as a stuck-open relief valve that result in a an uncontrolled loss of primary system coolant. The emergency core cooling system (ECCS) is designed to provide cooling water to the core terminating any heat up of the core and to provide long-term removal of the core decay heat. The containment provides the final barrier to prevent the release of fission products from the fuel. The suppression pool in the containment provides a heat sink and internal water source for core cooling. The containment heat removal systems transfer the decay heat to the ultimate heat sink. In addition to providing for core and containment cooling, the incorporation of the LOCA affected the structural design of the plant. The vessel and piping structures had to be designed to withstand the jet reaction and pipe whip forces from the broken pipe. Structures and systems had to be protected against the impingement of the jet from the broken pipe. Systems had to be designed to mitigate the hydrogen generated by the reaction of the zirconium fuel cladding with steam at high temperatures. The ECCS, containment, and supporting systems designed for mitigating the consequences of a LOCA have resulted in a network of safety system with the capacity and redundancy to handle a wide range of transient and accident events.

The BWR ECCS design has evolved along with the BWR product generations. The first generation of BWRs were designed without ECCS, but rather made use of highly reliable feedwater systems. With the increase in core power associated with the first commercial scale reactors (the BWR/2 design), concerns were raised with respect to providing adequate core cooling during a LOCA and preventing a core meltdown that would threaten containment integrity. In the late 1960s, the Report of the Advisory Task Force on Power Reactor Emergency Cooling led to the requirement that all nuclear plants must have ECCS [6]. Some forms of emergency core cooling, typically a core spray and high-pressure coolant injection (HPCI), were retrofit to the BWR/1s. The BWR/2s, BWR/3s, and BWR/4s were under design and construction during this time frame and a separate ECCS were incorporated into those reactor designs.

The BWR/2 recirculation loop design took water from the downcomer region and pumped it into the lower plenum region of the vessel. The limiting pipe break LOCA for the BWR/2 was in the recirculation discharge pipe where it attached to the vessel. A break at this location would leave a large rupture in the bottom of the system that would prevent reflooding the vessel. Therefore, it was decided that the best means of cooling the core would be with a spray system that would wet and cool the fuel from above. The core spray consisted of ring spargers around the periphery of the core, just above the top of the fuel bundles. The ring spargers do not interfere with the core flow during normal operation or fuel bundle movement during refueling. Two completely separate systems (electrical power, pumps, valves, piping, etc.) were provided for redundancy. Because the core spray system was designed for the large break LOCA, the core spray pumps are low-pressure, high-flow capacity pumps. These pumps cannot inject into the vessel at normal operating pressures. To address breaks that are so small that the vessel will not depressurize through the break, logic was added to some of the vessel pressure relief valves. When the logic detected the LOCA conditions, low reactor water level and high containment drywell pressure, and sensed that the core spray pumps were running, the logic would open the relief valves and depressurize the vessel, effectively making the small break into a large break. This logic is known as the automatic depressurization system (ADS). For the long-term recovery from the accident, provisions were made to flood the containment to a level above the top of the core. In addition to the ECCS network, the BWR/2 design includes ICs to provide decay heat removal from the vessel in the event that the main steam line isolation valves are closed and the

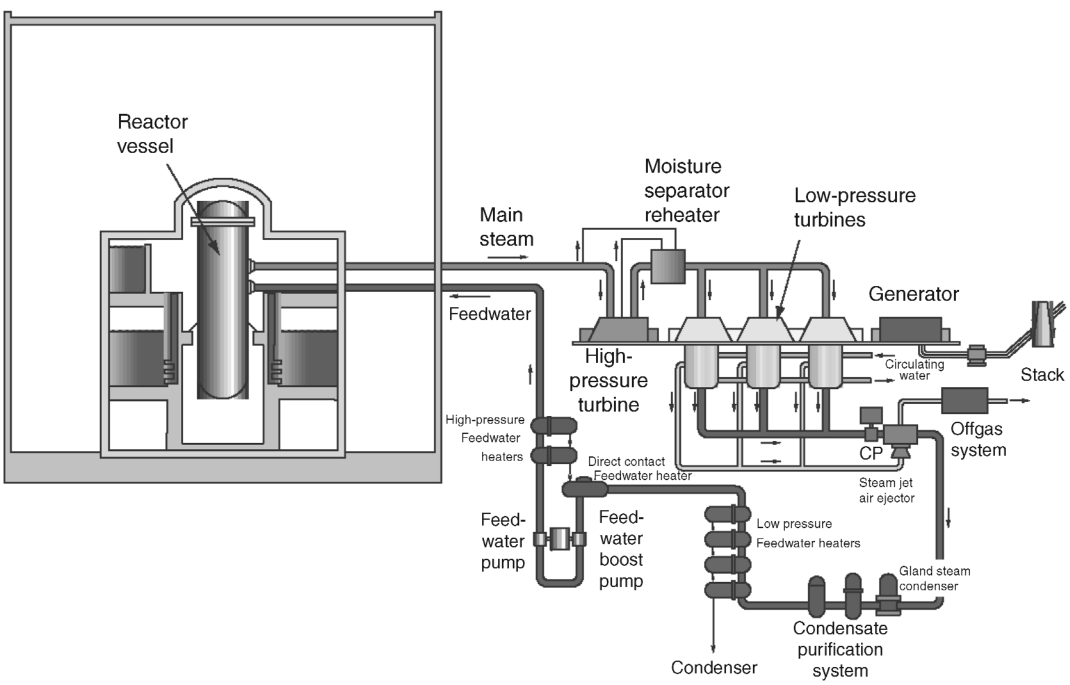

FIG. 58.2 ESBWR STEAM AND POWER CONVERSION SYSTEM [2]

main condenser is unavailable as a heat sink. The ICs are simply passive heat exchangers located in pools of water above the vessel and outside the containment. Steam from the reactor vessel is passed through the heat exchanger tubes, condensed, and the condensate is returned to the vessel. The water on the shell side of the heat exchange is boiled and vented to the atmosphere.

The BWR/2 plant design also marked the standardization of the pressure suppression containment design. In the pressure suppression containment design, steam released from the reactor vessel is directed into the suppression pool where it is condensed. The containment cooling system, which takes water from the suppression pool, passes it through a heat exchanger where it is cooled, then returns the cooled water to the containment, either back to the pool or through sprays in the drywell and suppression chamber airspace.

The ECCS network design used for the BWR/3 and BWR/4 product lines are similar. The two core spray systems and the ADS system were carried forward from the BWR/2. The jet pump design introduced in the BWR/3 eliminated the large recirculation pipe connection to the bottom head of the vessel. It now became possible to reflood the core region of the vessel up to the top of the jet pumps. The containment cooling system described above grew in scope and became the residual heat removal (RHR) system. The RHR system was connected to the reactor vessel to provide decay heat removal when the reactor is shut down. During the beginning portion of the accident, the flow from the four RHR pumps is directed into the recirculation discharge lines and injected through the jet pumps into the core shroud region of the vessel. This mode is called the low pressure coolant injection (LPCI) mode. Once the core is reflooded, the RHR pumps are realigned into the containment cooling mode. The two core spray systems are dedicated to core cooling. In addition, a turbine-driven large-capacity HPCI pump was added to the ECCS network. This pump provides inventory makeup for small breaks with the vessel at operating pressure. The improved core cooling provided by the ECCS network allowed larger plants with higher power cores to be designed. With the higher core powers, however, the size of the IC became impractical. In place of the ICs, the later BWR/3s and all the BWR/4s used a second nonsafety turbine-driven injection pump to provide inventory makeup when the vessel is isolated. The reactor core isolation cooling (RCIC) system is similar to the HPCI but smaller in capacity because it is sized to make up only the boiloff due to decay heat. The control valves and instrumentation for the HPCI and RCIC systems are powered from the station emergency batteries, allowing them to function in station blackout situations where no power is available from off-site or the station emergency diesel generators.

At the time the BWR/2, BWR/3, and BWR/4 ECCS were designed, there were no regulations defining the ECCS performance acceptance criteria. The design intent of the ECCS was to prevent the fuel cladding from melting and to keep the core in a coolable geometry during a LOCA. GE designed the ECCS for these plants to limit the peak cladding temperature below 2700 °F, which provided some margin to the melting point of approximately 3400 °F. In 1973, 10CFR50.46 was issued, which defined the ECCS performance acceptance criteria. In these new regulations, the peak cladding temperature was limited to 2200 °F. More sophisticated analytical models, combined with a transition in fuel design from a 7×7 lattice to an 8×8 lattice with the fuel rods operating at lower powers, allowed the plants to minimize the impact of the new regulations on the plant power output.

The BWR/5 product line was designed during the time frame when the ECCS performance issues were being raised. The ECCS network incorporated in the BWR/5 product line improved the performance and reliability of the ECCS over the BWR/3 and BWR/4 designs. The network consists of three electrical divisions, of which two divisions contain two ECCS pumps each, an RHR pump that can be used for LPCI injection or containment cooling and either a low-pressure core spray system or a second LPCI pump dedicated to vessel injection. The third electrical division has a high-pressure core spray (HPCS) system powered by a dedicated diesel generator, which replaces the turbine-driven HPCI. The HPCI turbine would trip off when the vessel depressurized below about 100 psi, whereas the electrically driven HPCS operates through the full vessel pressure range. Also, the HPCI injects water through the feedwater line into the vessel downcomer region. With a large recirculation line break, the water injected by the HPCI would be lost through the break and not reach the core region. The HPCS injects directly into the core region through one of the two core spray spargers. The LPCI flow path was rerouted from the recirculation discharge line to dedicated lines allowing the LPCI to inject directly into the core region inside the core shroud. In the earlier BWR/3 and BWR/4 designs, a break of the recirculation discharge line also disabled the injection flow path for the LPCI injecting into that division. The RCIC was retained to provide makeup when the vessel is isolated and for station blackout events. The BWR/5 design was carried forward into the BWR/6 product line with the capacities increased to accommodate the higher BWR/6 core power levels. The ECCS network design was able to meet the requirements imposed by 10CFR50.46 without any impact on plant operations.

The internal recirculation pumps in the ABWR eliminated the large external recirculation loop piping. The ABWR was the first BWR designed after the adoption of the 10CFR50.46 limits. With the elimination of the large pipes attached to the lower section of the vessel, it was now possible to design the ECCS so that the core would remain covered during the design basis LOCA, thus assuring adequate core cooling throughout the event. It was possible to significantly downsize ECCS equipment as a result of eliminating large vessel nozzles below the top of the core. Capacity requirements are sized based on operating requirements – transient response and shutdown cooling – rather than on the need for large reflood capability. Inside the reactor vessel, core spray spargers were eliminated, since no postulated LOCA would lead to core uncovery. For transient response, the initiation water levels for RCIC and the high-pressure core flooder were separated so that there is reduced duty on the equipment relative to earlier BWRs. There are three complete shutdown cooling loops, including dedicated vessel nozzles. Complex operating modes of the RHR systems, such as steam condensing, were eliminated. Finally, heat removal, in addition to core injection, was automated so that the operator no longer needs to choose which mode to perform during transients and accidents.

Like the ABWR, the ESBWR is designed to keep the core covered and cooled during a LOCA. However, the ESBWR core and containment cooling systems represent a radical departure from those of the earlier BWR product lines in that the cooling systems are passive and do not rely on electrically driven pumps. The ESBWR passive safety systems are discussed in more detail in Section 58.3.3.

58.2.4 Containment Design

The containment is essentially a pressure vessel housing the reactor that provides a safety barrier to protect personnel and the public in the event of a reactor accident. Figure 58.3 illustrates the evolution of

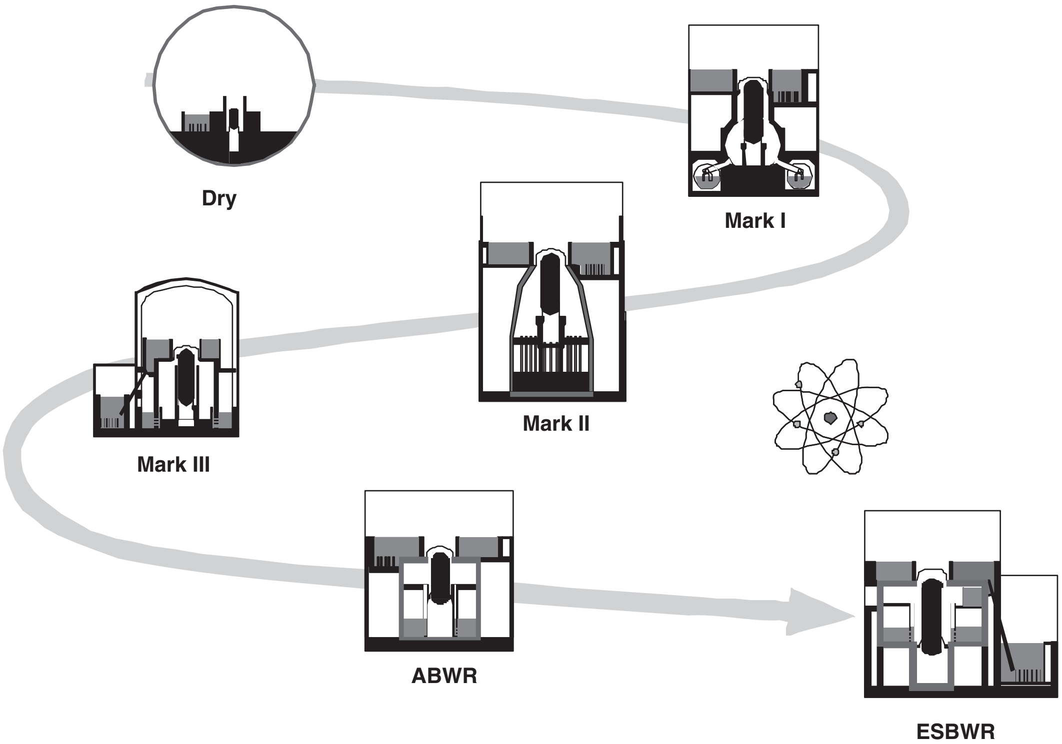

FIG. 58.3 EVOLUTION OF THE BWR CONTAINMENT DESIGN [2]

the BWR containment from the earliest versions to today's ESBWR design. Most of the initial BWR/1 plants used spherical dry containments. The containment vessel was sized to be able to hold the sensible energy content of the steam and hot water in the reactor coolant system and limit the pressure rise to within the design value of the vessel. Larger reactor sizes could be accommodated by increasing the volume of the containment vessel and by increasing the pressure rating of the vessel. However, because of the higher energy content in the BWR vessel as compared to a similar PWR, dry containments could not be designed that were economically competitive with PWRs. Therefore, GE selected the pressure suppression containment for the standard BWR plants [7].

Table 58.3 provides a comparison of the key features of the pressure suppression containments used in the BWR plants. In a pressure suppression containment, the reactor and primary system piping are contained in one chamber, the drywell, and the suppression pool is contained in a separate chamber, the wetwell. A system of vents provides a flow path from the drywell to the wetwell, with the vents discharging below the surface of the suppression pool. In the event of a LOCA, the steam and water from the vessel, as well as the air in the drywell, are driven through the vents into the suppression pool where the water in the pool quenches the steam. The advantages of a pressure suppression containment are as follows:

- High heat capacity
- Lower design pressure
- Superior ability to accommodate rapid depressurization
- The pool has the ability to filter and retain fission products
- A large source of readily available makeup water in the case of accidents
- Simplified, compact design

The Mark I containment was the first of the pressure suppression containment designs. The Mark I design has a characteristic light bulb configuration for the drywell, surrounded by a steel torus that houses a large water pressure suppression pool. The light bulb design came from the need to have a removable closure at the top for reactor servicing and refueling and sufficient room at the bottom for the recirculation piping. The torus design provided a large surface area for the vents. The structural analyses of the drywell and the torus were also simplified because closed-form solutions were available for these simple shapes (the Mark I design was developed in an era of limited computing power). The drawbacks of the Mark I design was the difficulty of its construction and the construction of the reactor building around it and the limited room inside the drywell for construction of the primary system piping and maintenance of components. The Mark I containment was used with the BWR/2, BWR/3, and BWR/4 product lines.

The Mark II over-under configuration was designed to address the construction issues associated with the Mark I design. The main advantages of the Mark II design were (1) more volume in the drywell to allow better access to the steam and ECCS piping, (2) simpler vent configuration using straight pipes, (3) the potential to use different construction materials, and (4) a smaller reactor building. The Mark II containment design was introduced with the BWR/5 product line but was also used for a couple of the late BWR/4 plants.

The Mark III containment design, introduced with the BWR/6 product line, represented a major improvement in simplicity. Its containment structure is a right circular cylinder that is easy to construct, and provides access to equipment and ample space for maintenance activities. The large volume allowed a reduction in the design pressure. Other features of the Mark III include horizontal

TABLE 58.3 COMPARISON OF KEY FEATURES OF GE BWR CONTAINMENTS

Parameter	Mark I	Mark II	Mark III	ABWR	ESBWR
Power level (MWe)	1100	1100	1220	1371	1600
Reactor pressure vessel inside diameter (m)	6.4	6.4	6.0	7.1	7.1
Drywell					
Free volume (m^3)	4672	7872	7929	7350	7206
Design pressure (MPa)	0.43	0.31	0.17	0.31	0.31
Wetwell					
Free volume (m^3)	4834	5318	32904	5960	5467
Water volume (m^3)	3480	3268	4332	3580	4383
Design pressure (MPa)	0.43	0.31	0.10	0.31	0.31
Vents					
Orientation	Vertical	Vertical	Horizontal	Horizontal	Horizontal
Size (m)	0.6	0.6	0.7	0.7	0.7
Number	76	70	117	30	30

vents to reduce overall loss-of-coolant (LOCA) dynamic loads and a freestanding all-steel structure to ensure leak tightness.

The ABWR containment is significantly smaller than the Mark III containment because the elimination of the recirculation loops translates into a significantly more compact containment and reactor building. The structure itself is made of reinforced concrete with a steel liner from which it derives its name – reinforced concrete containment vessel (RCCV). The ESBWR containment is similar in construction to the ABWR, but slightly larger to accommodate the passive ECCS systems.

58.3 KEY FEATURES OF ESBWR

58.3.1 Natural Circulation Design

Natural circulation in BWRs is a proven technology [8, 9]. Some of the early GE BWRs employed natural circulation. These were small plants (e.g., Dodewaard at 183 MWt and Humboldt Bay at 165 MWt), but they clearly demonstrated the feasibility of the BWR and provided valuable operating data and experience. GE moved to forced circulation plants to achieve higher power ratings in a compact pressure vessel. Pressure vessel fabrication capability at the time was a factor in this decision. Now, after several decades, GE is returning to natural circulation for the ESBWR.

Natural circulation provides major simplification by removal of the recirculation pumps and associated piping and heat exchangers and controls. It is also synergistic with two other requirements that GE considered to be important in the design of a new reactor: large safety margins with a very reliable passive ECCS and avoidance of safety/relief valve (SRV) opening for pressurization transients such as turbine trips or main steam line isolation events. Both of these features need a tall pressure vessel with large water volumes. The tall vessel leads to enhanced natural circulation flow, so the natural circulation capability comes with no additional cost.

The ESBWR builds on the design features of operating BWRs. Figure 58.4 shows a cutaway of the ESBWR RPV. Most components in the ESBWR are standard BWR components that have been operating in the field for years (steam separators, control rods and guide tubes, core support structure, etc.). The core consists of conventional BWR fuel bundles, shortened from 12 ft. to 10 ft. to improve pressure drop and stability characteristics. The absence of hardware in the downcomer (jet pumps or internal pumps) reduces flow losses and further enhances natural circulation. The main difference is the taller reactor vessel with the addition of a partitioned chimney above the core and a correspondingly taller downcomer annulus. The fluid in the taller downcomer provides the additional driving head for natural circulation flow through the core, as well as a large water inventory for a LOCA. Steam in the chimney also provides a cushion to dampen void collapse in the core during pressurization transients, leading to a softer response with no SRV discharges.

58.3.2 Operating Domain

Valuable operating experience was gained from the early natural circulation BWRs. It was demonstrated that BWRs could operate in natural circulation without problems. The plants were extremely stable and presented no unusual characteristics relative to noise in the instrumentation. Power was raised by control rod withdrawal. The ESBWR will also adjust output using control rods, but with electrically driven CRDs that move slower and have finer positioning capability than the locking piston design for conventional BWRs. Because of its size, ESBWR will not normally be operated in a load follow mode. The FMCRDs can accommodate a duty corresponding to daily load following cycles for 10 years. Variable feedwater temperature control has been added to provide a second means of reactivity control in addition to the fine motion CRDs.

The present-day BWRs can operate at about 50% of rated power in natural circulation; however, stability considerations prevent

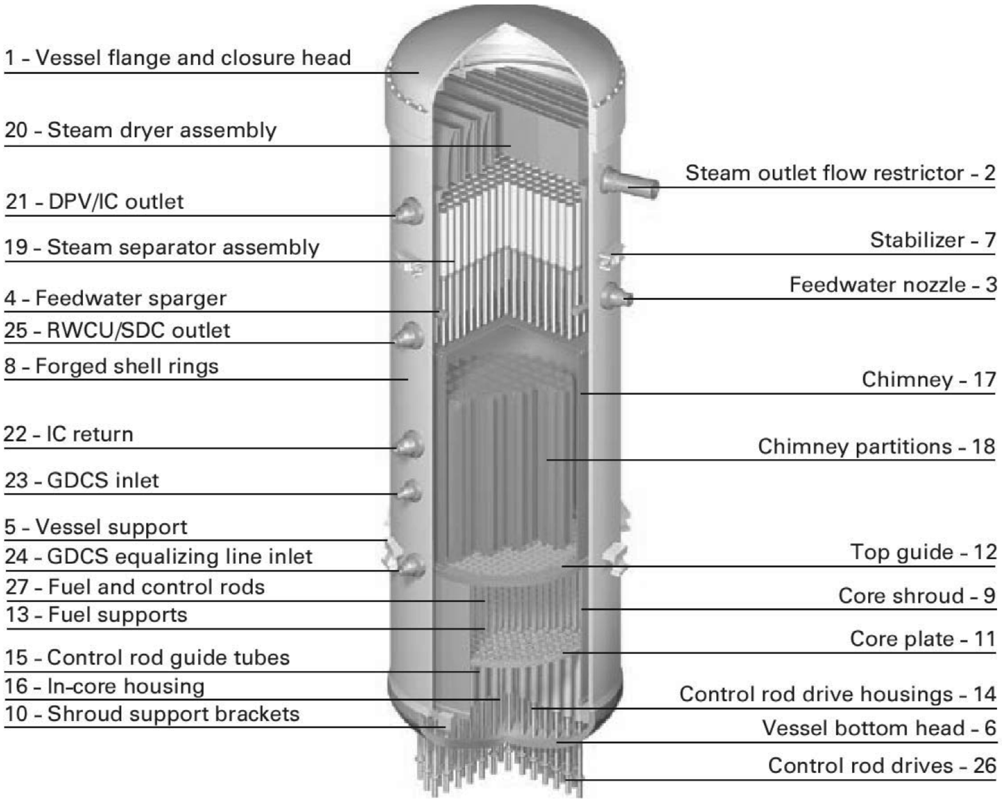

FIG. 58.4 ESBWR REACTOR ASSEMBLY [2]

steady operation in this region. There have been recirculation pump trip events in operating plants, which led to a natural circulation state at around 50% of rated power. The operating conditions in the present-day BWRs (power, flow, power distribution) in natural circulation following the pump trip were well predicted by the calculational models used for ESBWR performance analysis.

The operating parameters for the ESBWR, such as the power density, steam quality, void fraction, and void coefficient are within the range of operating plant data. Figure 58.5 shows a comparison of the ESBWR power–flow operating map with those of operating BWRs. This figure is based on the power per bundle and flow per bundle so that a meaningful comparison can be made. The power per bundle and flow per bundle for the ESBWR are both lower than for a modern jet pump plant at rated operating conditions, but the ratio of power to flow is similar to that for an uprated BWR at Maximum Extended Load Line Limit Analysis Plus (MELLLA+) conditions. This means that the core exit steam quality (ratio of steam flow to core flow) is also similar.

58.3.3 Passive Safety Features

The ESBWR safety systems design incorporates four redundant and independent divisions of the passive GDCS, the ADS and a PCCS. These systems are shown in Fig. 58.6. Heat removal and inventory addition are also provided by the isolation condenser system (ICS) and the standby liquid control system (SLCS).

The RPV has no external recirculation loops or large pipe nozzles below the top of the core region. This, together with a high capacity ADS, allowed the incorporation of an ECCS driven solely by gravity, not needing any pumps. The water source needed for the ECCS function is stored in the containment upper drywell, with sufficient water to insure core coverage to 1 m above the top of active fuel as well as flooding the lower drywell.

The PCCS heat exchangers are located above and immediately outside of containment. There is sufficient water in the external pools to remove decay heat for at least 72 h following a postulated design basis accident, and provisions exist for external makeup beyond that, if necessary.

As a result of these simplifications in the ESBWR safety systems, there is an increase in the calculated safety performance margin of the ESBWR over earlier BWRs. This has been confirmed by a Probabilistic Risk Assessment (PRA) for the ESBWR, which shows that the ESBWR is a calculated factor of about 5 lower than ABWR and 50 better than BWR/6 in avoiding possible core damage from degraded events.

58.3.3.1 Gravity-Driven Core Cooling System The GDCS is composed **of** four divisions. A single division of the GDCS consists of three independent subsystems: a short-term cooling (injection) system, a long-term cooling (equalizing) system, and a deluge line. The short-term and long-term systems provide cooling water under force of gravity to replace RPV water inventory lost during a postulated LOCA and subsequent decay heat boiloff. The deluge line connects the GDCS pool to the lower drywell. A schematic of the GDCS is shown in Fig. 58.7.

Each division of the GDCS injection system consists of one 200-mm pipe exiting from the GDCS pool. A 100-mm deluge line branches off and is terminated with three 50-mm squib valves and deluge line tailpipes to flood the lower drywell. The injection line

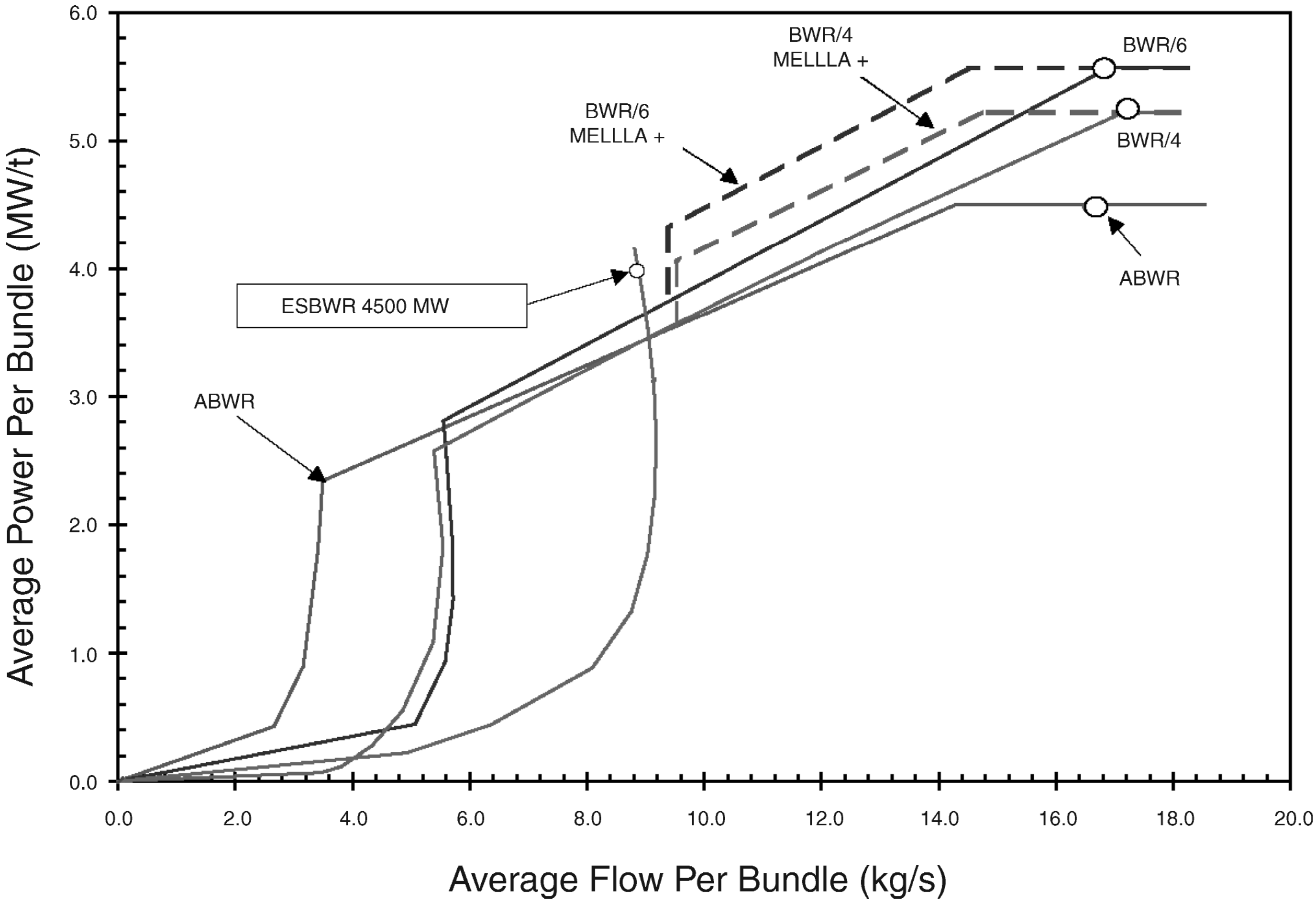

FIG. 58.5 COMPARISON OF ESBWR OPERATING POWER/FLOW MAP WITH OPERATING BWRs [2]

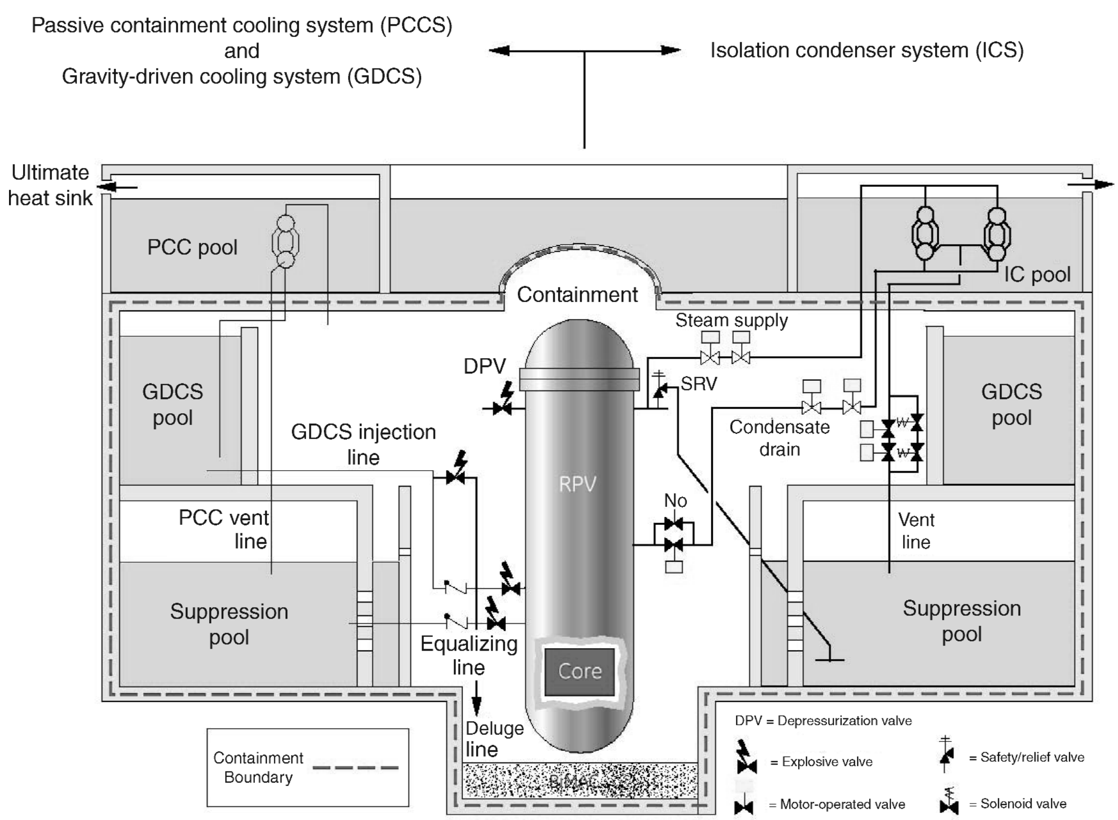

FIG. 58.6 ESBWR KEY SAFETY SYSTEMS [2]

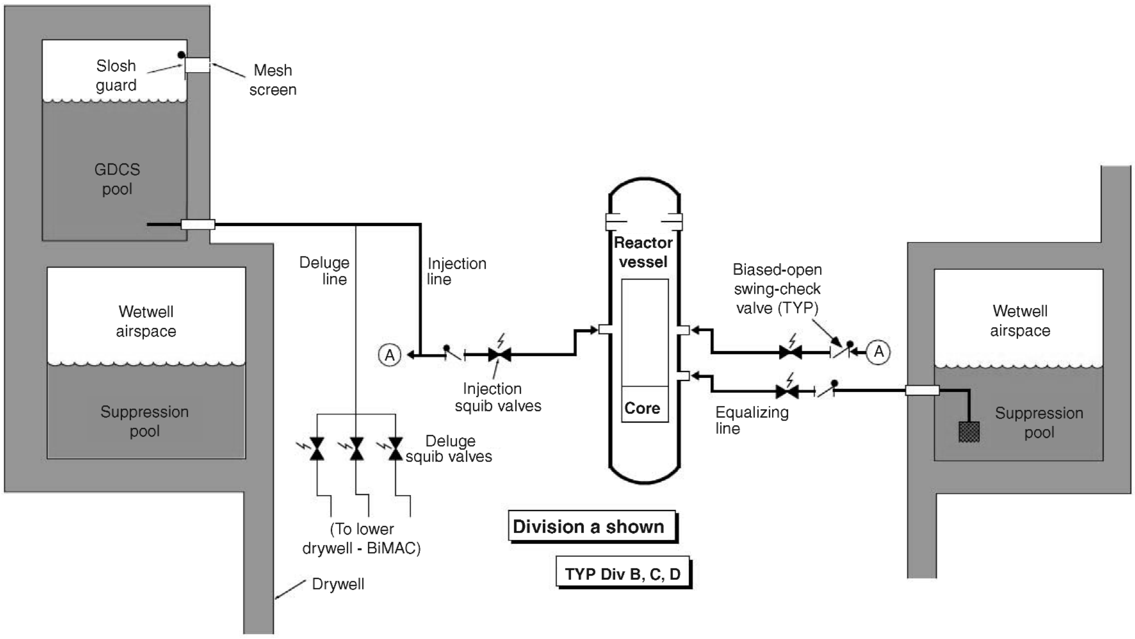

FIG. 58.7 ESBWR GRAVITY-DRIVEN COOLING SYSTEM [2]

continues after the deluge line connection from the upper drywell region through the drywell annulus where the line branches into two 150-mm branch lines, each containing a biased-open check valve and a squib valve.

Each division of the long-term system consists of one 150-mm equalizing line with a check valve and a squib valve, routed between the suppression pool and the RPV. All piping is of stainless steel and rated for reactor pressure and temperature. The RPV injection line nozzles and the equalizing line nozzles all contain integral flow limiters.

In the injection lines and the equalizing lines, there exists a biased-open check valve located upstream of the squib-actuated valve. The GDCS squib valves are gas propellant-type shear valves that are normally closed and which open when a pyrotechnic booster charge is ignited. During normal reactor operation, the squib valve is designed to provide zero leakage. Once the squib valve is actuated, it provides a permanent open flow path to the vessel.

The check valves mitigate the consequences of spurious GDCS squib valve operation and minimize the loss of RPV inventory after the squib valves are actuated and the vessel pressure is still higher than the GDCS pool pressure including (its gravity head). Once the vessel has depressurized below the GDCS pool surface pressure and its gravity head, the differential pressure opens the check valve and allow water to begin flowing into the vessel.

The GDCS deluge lines provide a means of flooding the lower drywell region with GDCS pool water in the event of a postulated core melt sequence that causes failure of the lower vessel head and allows the molten fuel to reach the lower drywell floor. A core melt sequence would result from a common mode failure of the short-term and long-term systems, which prevents them from performing their intended function. Deluge line flow is initiated by thermocouples, which sense high lower drywell region basemat temperature indicative of molten fuel on the lower drywell floor. Squib-type valves in the deluge lines are actuated upon detection of high basemat temperatures. The deluge lines do not require the actuation of squib-actuated valves on the injection lines of the GDCS piping to perform their function.

The deluge valves are opened based on very high temperatures in the lower drywell, indicative of a severe accident. Once the deluge valve is actuated, it provides a permanent open flow path from the GDCS pools to the lower drywell region. Flow then drains to the lower drywell via permanently open drywell lines. This supports the BiMAC core catcher function.

The GDCS check valves remain partially open when zero differential pressure exists across the valve. This is to minimize the potential for sticking in the closed position during long periods of nonuse.

Suppression pool equalization lines have an intake strainer to prevent the entry of debris material into the system that might be carried into the pool during a large break LOCA. The GDCS pool airspace opening to DW is covered by a mesh screen or equivalent to prevent debris from entering pool and potentially blocking the coolant flow through the fuel. A slosh guard is added to the opening to minimize any sloshing of GDCS pool water into the drywell following dynamic events.

The GDCS equalizing lines perform the RPV inventory control function in the long term. By closing the loop between the suppression pool and the RPV, liquid inventory that is transferred to the suppression pool either by PCCS condensation shortfall or by steam condensation in the drywell (which eventually spills back to the suppression pool) can be added back to the RPV.

58.3.3.2 Passive Containment Cooling System The PCCS maintains the containment within its pressure limits for DBAs. The system is designed as a passive system with no components that must actively function, and it is also designed for conditions that equal or exceed the upper limits of containment severe accident capability. The PCCS consists of six low-pressure, totally independent loops, each containing a steam condenser (passive containment cooling condenser), as shown in Fig. 58.8. Each PCCS

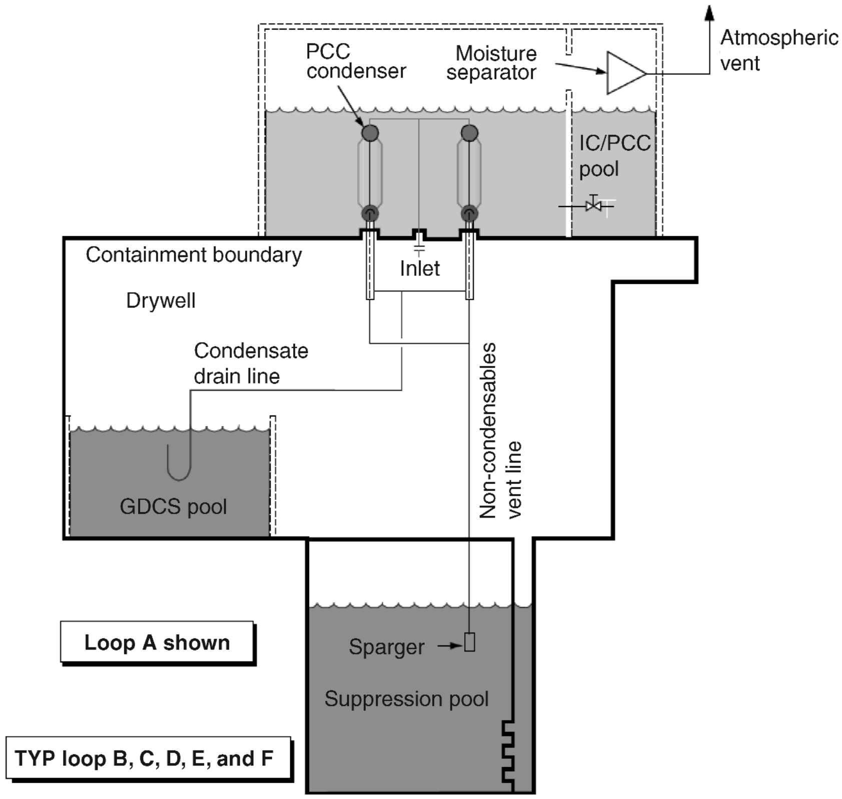

FIG. 58.8 ESBWR PASSIVE CONTAINMENT COOLING SYSTEM [2]

condenser loop is designed for 11-MWt capacity and is made of two identical modules. Together with the pressure suppression containment, the PCCS condensers limit containment pressure to less than its design pressure for at least 72 h after a LOCA without makeup to the IC/PCC pool and beyond 72 h with pool makeup. The PCCS condensers are located in a large pool (IC/PCC pool) positioned above, and outside, the ESBWR containment (DW).

Each PCCS condenser loop is configured as follows. A central steam supply pipe is provided, which is open to the containment at its lower end and it feeds two horizontal headers through two branch pipes at its upper end. Steam is condensed inside vertical tubes and the condensate is collected in two lower headers. The vent and drain lines from each lower header are routed to the DW through a single containment penetration per condenser module as shown in the diagram. The condensate drains into an annular duct around the vent pipe and then flows in a line that connects to a large common drain line, which also receives flow from the other header, ending in a GDCS pool.

The noncondensable vent line is the pathway by which drywell noncondensables are transferred to the wetwell. This ensures a low noncondensable concentration in the steam in the condenser, necessary for good heat transfer. During the periods in which PCCS heat removal is less than decay heat, excess steam also flows to the suppression pool via this pathway.

The PCCS loops receive a steam–gas mixture supply directly from the DW. The PCCS loops are initially driven by the pressure between the containment DW and the suppression pool during a LOCA and then by gravity drainage of steam condensed in the tubes, so they require no sensing, control, logic, or power-actuated devices to function. The PCCS loops are an extension of the safety-related containment and do not have isolation valves.

Spectacle flanges are included in the drain line and in the vent line to conduct postmaintenance leakage tests separately from Type A containment leakage tests. Located on the drain line and submerged in the GDCS pool, just upstream of the discharge point, is a loop seal. It prevents backflow of steam and gas mixture from the DW to the vent line, which would otherwise short circuit the flow through the PCCS condenser to the vent line. It also provides long-term operational assurance that the PCCS condenser is fed via the steam supply line.

Each PCCS condenser is located in a subcompartment of the IC/PCC pool, and all pool subcompartments communicate at their lower ends to enable full use of the collective water inventory independent of the operational status of any given IC/PCCS subloop. A valve is provided at the bottom of each PCC subcompartment that can be closed so that the subcompartment can be emptied of water to allow PCCS condenser maintenance.

Pool water can heat up to about 101 °C (214 °F); steam formed, being nonradioactive and having a slight positive pressure relative to station ambient, vents from the steam space above each PCCS condenser where it is released to the atmosphere through large-diameter discharge vents. A moisture separator is installed at the entrance to the discharge vent lines to preclude excessive moisture carryover and loss of IC/PCC pool water. IC/PCC pool makeup clean water supply for replenishing level is normally provided from the makeup water system.

Level control is accomplished by using an air-operated valve in the makeup water supply line. The valve opening and closing is controlled by water level signal sent by a level transmitter sensing water level in the IC/PCC pool. Cooling and cleanup of IC/PCC pool water is performed by the fuel and auxiliary pools cooling system (FAPCS). The FAPCS provides safety-related, dedicated makeup piping, independent of any other piping, which provides an attachment connection at grade elevation in the station yard outside the reactor building, whereby a post-LOCA water supply can be connected. There have been extensive qualification tests of the PCCS, including full-scale component tests and full height scaled integral tests.

58.3.3.3 Standby Liquid Control System The SLCS provides a backup method to bring the nuclear reactor to subcriticality and to maintain subcriticality as the reactor cools. The system makes possible an orderly and safe shutdown in the event that not enough control rods can be inserted into the reactor core to accomplish shutdown in the normal manner. The SLCS is sized to counteract the positive reactivity effect of shutting down from rated power to cold shutdown condition. It also adds additional inventory to the RPV after confirmation of a LOCA.

The SLCS is automatically initiated in case of signals indicative of LOCA or anticipated trips without scram (ATWS). It can also be manually initiated from the main control room to inject the neutron-absorbing solution into the reactor.

The SLCS is a two-division passive system using pressurized accumulators to inject borated water rapidly and directly into the bypass area of the core. Each division is of 50% capacity. Injection will take place after either of two squib valves in each division fires upon actuation signal from the SSLC. Figure 58.9 illustrates the SLCS configuration.

In addition to the accumulators and injection valves, supporting nonsafety grade equipment includes a high-pressure nitrogen charging system for pressurization and to make up for losses and a mixing and boron solution makeup system. The boron absorbs thermal neutrons and thereby terminates the nuclear fission chain reaction in the fuel. The specified neutron absorber solution is sodium pentaborate using 94% of the isotope B10 at a concentration of 12.5%. This combination not only minimizes the quantity of liquid to be injected, but also assures no auxiliary heating is needed to prevent precipitation of the sodium pentaborate out of solution in the accumulator and piping. At all times, when it is possible to make the reactor critical, the SLCS will be able to deliver enough sodium pentaborate solution into the reactor to assure reactor shutdown.

Upon completion of injecting the boron solution, redundant accumulator level measurement instrumentation using two out of four logic closes the injection line shut-off valve in each SLCS division. Closure of these valves prevents injection of nitrogen from the accumulator into the reactor vessel that could interfere with ICS operation or cause additional containment pressurization. As a backup, the accumulator vent valves are also opened at the same time.

58.3.3.4 Isolation Condenser System The primary function of the ICS is to limit reactor pressure and prevent SRV operation following an isolation of the main steam lines. The ICS, together with the water stored in the RPV, conserves sufficient reactor coolant volumes to avoid automatic depressurization caused by low reactor water level. The ICS removes excess sensible and core decay heat from the reactor, in a passive way and with minimal loss of coolant inventory from the reactor, when the normal heat removal system is unavailable. The ICS is designed as a safety-related system to remove reactor decay heat following reactor shutdown and isolation. It also prevents unnecessary reactor depressurization and operation of ECCS, which can also perform this function.

The ICS consists of four totally independent trains, each containing an IC that condenses steam on the tube side and transfers heat to a large IC/PCCS pool positioned immediately outside the containment, which is vented to the atmosphere as shown in the ICS schematic (Fig. 58.10). The IC, connected by piping to the RPV, is placed at an elevation above the source of steam (vessel) and, when the steam is condensed, the condensate is returned to the vessel via a condensate return pipe. The steam side connection between the vessel and the IC is normally open and the condensate line is normally closed. This allows the IC and drain piping to fill with condensate, which is maintained at a subcooled temperature by the pool water during normal reactor operation. The IC is brought into operation by opening

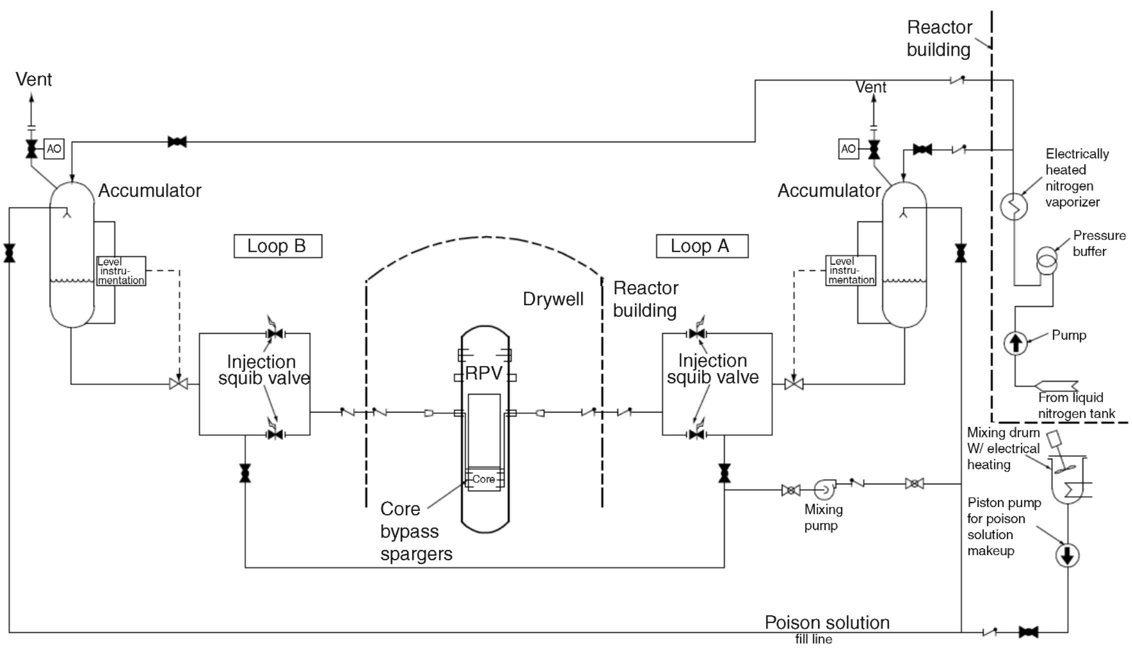

FIG. 58.9 ESBWR STANDBY LIQUID CONTROL SYSTEM [2]

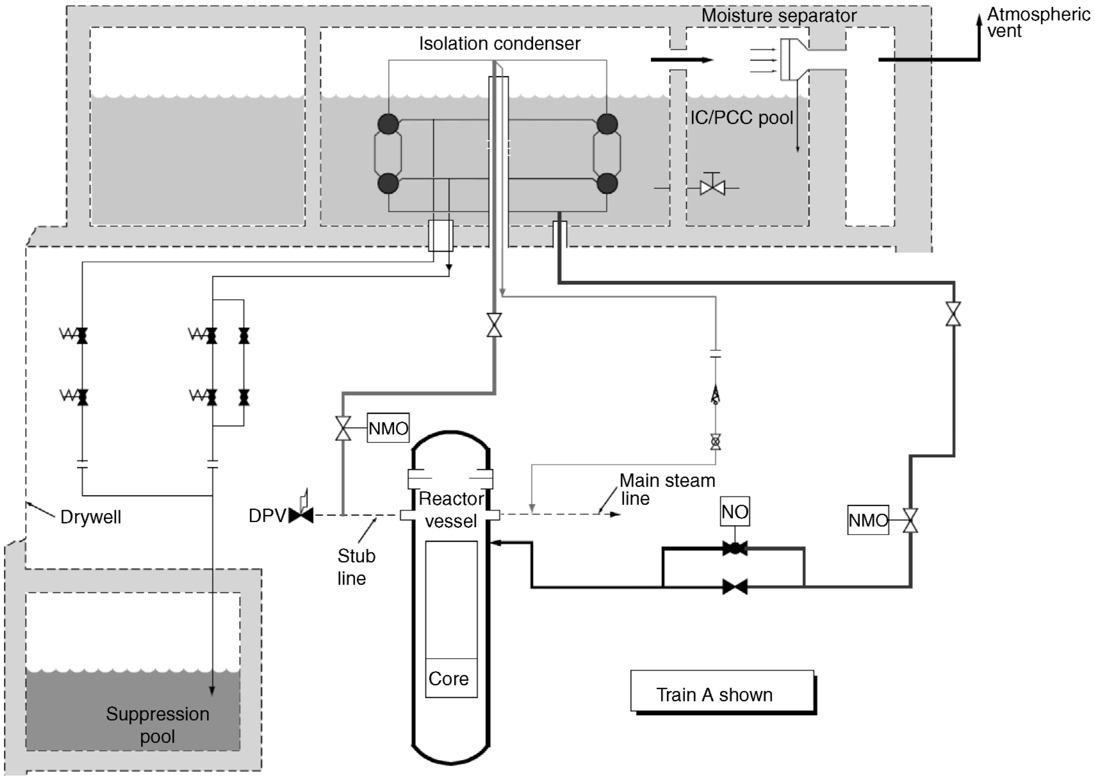

FIG. 58.10 ESBWR ISOLATION CONDENSER SYSTEM [2]

condensate return valves and draining the condensate to the reactor, thus causing steam from the reactor to fill the tubes that transfer heat to the cooler pool water. Each IC is made of two identical modules.

The steam supply line (properly insulated and enclosed in a guard pipe that penetrates the containment roof slab) is vertical and feeds two horizontal headers through four branch pipes. Each pipe is provided with a built-in flow limiter, sized to allow natural circulation operation of the IC at its maximum heat transfer capacity while addressing the concern of IC breaks downstream of the steam supply pipe. Steam is condensed inside vertical tubes and condensate is collected in two lower headers. Two pipes, one from each lower header, take the condensate to the common drain line that vertically penetrates the containment roof slab.

A vent line is provided for both upper and lower headers to remove the noncondensable gases away from the unit during the IC operation. The vent lines are routed to the containment through a single penetration.

A purge line is provided to assure that during the normal plant operation (IC system standby conditions), the excess of hydrogen (from hydrogen water chemistry control additions) or air from the feedwater does not accumulate in the IC steam supply line, thus assuring that the IC tubes are not be blanketed with noncondensables when the system is first started. The purge line penetrates the containment roof slab.

Containment isolation valves are provided on the steam supply piping and the condensate return piping.

Located on the condensate return piping just upstream of the reactor entry point is a loop seal and a parallel-connected pair of valves: (1) a condensate return valve (motor-operated, fail as is) and (2) a condensate return bypass valve (nitrogen piston-operated, fail open). These two valves are closed during normal station power operations. Because the steam supply line valves are normally open, condensate forms in the IC and develops a level up to the steam distributor, above the upper headers. To bring an IC into operation, the motor-operated condensate return valve is opened, whereupon the standing condensate drains into the reactor and the steam—water interface in the IC tube bundle moves downward below the lower headers to a point in the main condensate return line. The fail-open nitrogen piston-operated condensate return bypass valve opens if power is lost or on low reactor water level signal.

The loop seal assures that condensate valves do not have hot water on one side of the disk and ambient temperature water on the other side during normal plant operation, thus affecting leakage during system standby conditions. Furthermore, the loop seal assures that steam continues to enter the IC preferentially through the steam riser, irrespective of water level inside the reactor, and does not move countercurrent back up the condensate return line.

During ICS normal operation, any noncondensable gases collected in the IC are vented from the IC top and bottom headers to the suppression pool. During ICS standby operation, discharge of noncondensable gases is accomplished by a purge line that takes a small stream of gas from the top of the IC and vents it downstream of the RPV on the main steam line upstream of the MSIVs.

Radiation monitors are provided in the IC/PCC pool steam atmospheric exhaust passages for each IC loop. The radiation monitors are used to detect IC loop leakage outside the containment and cause either alarms or automatic isolation of a leaking IC. The IC has undergone engineering development testing using a prototype to demonstrate the proper operability, reliability, and heat removal capability of the design.

58.4 MATERIALS, FABRICATION, AND APPLICABLE ASME CODE EDITION

This section describes the thought process and the technical basis used in selecting materials and fabrication processes used in GE-designed ABWR and ESBWR. Additionally, the ASME Code edition applicability is briefly discussed.

58.4.1 RPV Design

Figure 58.4 shows the ESBWR reactor assembly. The materials and fabrication techniques mostly are common for ABWR and ESBWR. The RPV closure head is elliptical in shape and is fabricated of low alloy steel, as per ASME SA-508, Grade 3, Class 1. It is secured to the RPV by 80 sets of fasteners (studs and nuts). The studs are fabricated from SA-540 B23 or B24. The nuts are tightened in groups of (typically) four at a time, using an automatic or semiautomatic four-stud tensioner device.

There are three feedwater nozzles for each of the two feedwater lines, which utilize double thermal sleeves welded to the nozzle. The double thermal sleeve protects the vessel nozzle inner blend radius from the effects of high frequency thermal cycling. A schematic of the feedwater nozzle is shown in Fig. 58.11.

The vessel supports are of the sliding block-type geometry and are provided at a number of positions around the periphery of the vessel. Multiple vessel supports along with the corresponding pedestal RPV support brackets provide

- openings to permit water to pass from the upper to lower drywell
- access for ISI of the bottom head weld

The vessel bottom head consists of a spherical bottom cap, made from a single forging, extending to the toroidal knuckle between the head and the vessel cylinder and encompassing the CRD penetrations. With a thickness of approximately 260 mm, the bottom head meets the ASME Code allowables for the specified design loads. The main advantage of using a single forging for the bottom head is that it eliminates all RPV welds within the CRD pattern, thus reducing future ASME Section XI ISI requirements.

Both the ABWR and ESBWR RPVs utilize rings forged from low alloy steel, adjacent to and below the core belt line region. The flanges and large nozzles are also low alloy steel. The shell rings above the core belt line region and the RPV closure head are made from low alloy steel forgings or plate as per ASME SA-533, Type B, Class1. The required reference nil ductility temperature, RT_{NDT}, of the vessel material is −20 °C. Figure 58.12 shows one of the forged RPV shell rings during fabrication of an ABWR vessel.

58.4.2 Materials Selection and Water Chemistry Controls

From the experience gained from the operating BWR fleet, the understanding of both the materials of construction and the degradation mechanisms has increased significantly. This understanding

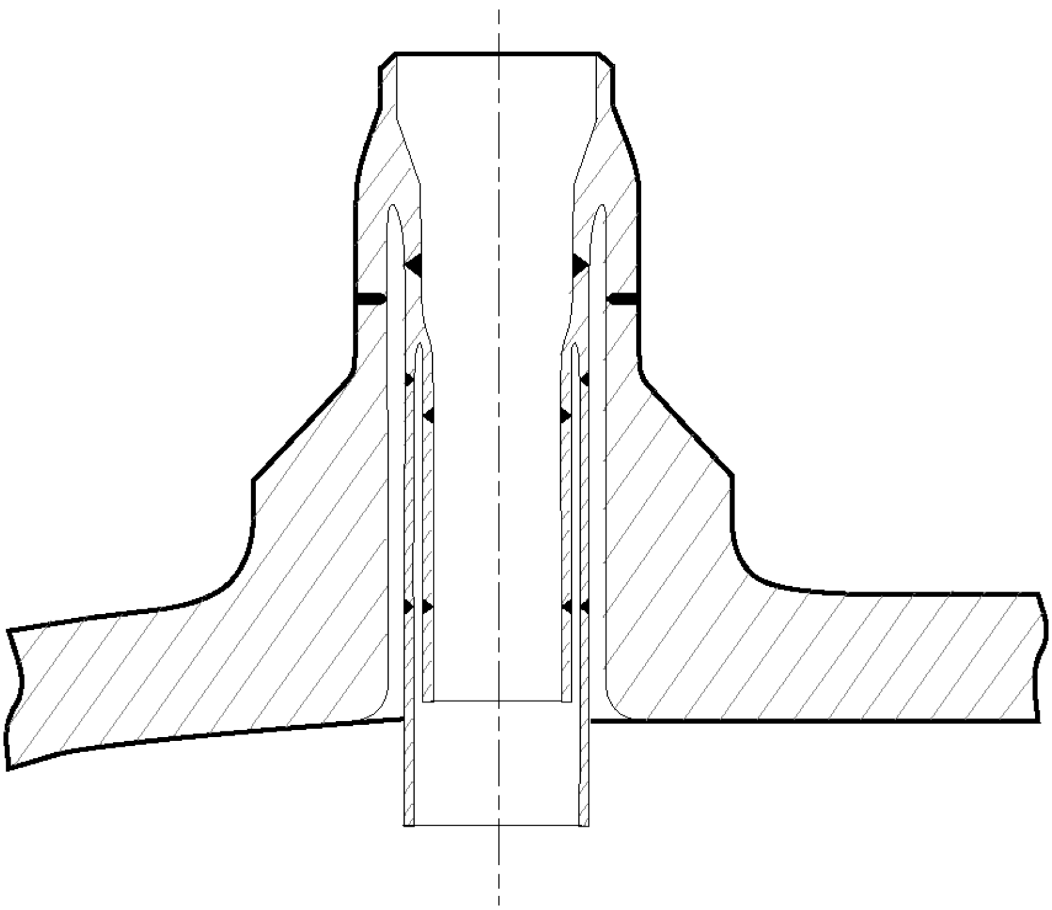

FIG. 58.11 ABWR/ESBWR RPV FEEDWATER NOZZLE [1, 2]

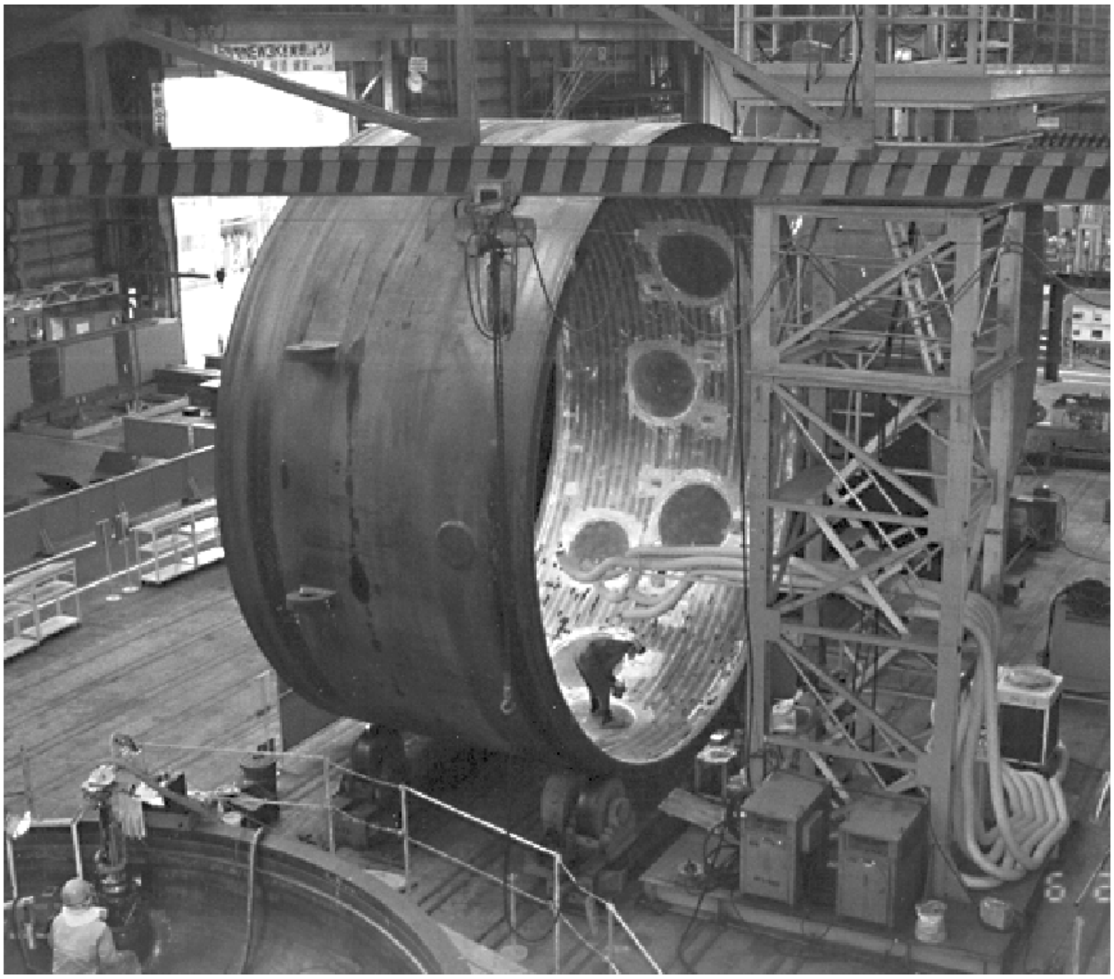

FIG. 58.12 ABWR RPV FORGED STEEL RING [1]

has been used to make ABWR/ESBWR materials selection. The ABWR/ESBWR makes use of austenitic stainless steels and nickel-base alloys as well as low alloy and carbon steel for all the major components. The plant design takes into account the potential effects of stress corrosion cracking (SCC), irradiation embrittlement, erosion/corrosion, and radiation buildup in the careful application of these materials. This application of materials is made in conjunction with good fabrication practice and strict operating control of the coolant environment chemistry to minimize the potential for long-term degradation and radiation buildup. A more detailed explanation of each area is given in the next section.

58.4.2.1 Austenitic Stainless Steel Materials for Internals and Components Austenitic stainless steels, because of high general corrosion resistance, high toughness, ease of fabrication, and acceptable strength, have been commonly used for reactor internals and some piping systems. In the ABWR, the primary consideration in the use of these materials is the development of resistance to intergranular stress corrosion cracking (IGSCC) both in low and high fluence locations. Based on field and laboratory efforts, this resistance is tied to the control of the material's composition, control of the fabrication processes, and control of the coolant environment.

The extensive understanding has now thoroughly demonstrated that reduction of carbon content in austenitic stainless steel reduces susceptibility to IGSCC. Three main austenitic materials are being used: Type 316NG/316L wrought material, CF3M cast materials, and XM-19 high strength stainless steel.

For the majority of the reactor internal structures, resistance to sensitization for the ABWR is achieved by using a special Type 316NG (nuclear grade). This alloy has carbon restricted to a maximum 0.020% by weight to prevent sensitization, has nitrogen additions up to a maximum of 0.12% to maintain the desired strength levels, and has specific fabrication and processing controls to increase the resistance to any crack initiation. Alternately, where design does not need the mechanical strength of Type 316, Type 316L is used with the same carbon limit and controls on fabrication processes.

Types 308L/316L weld metal and CF3M austenitic stainless steel castings are also used in the reactor as well as for many of the complex-shaped components such as valve body castings. These alloys have a duplex microstructure that has been found to have very high resistance to IGSCC. Their resistance is maximized by again limiting carbon content to less than 0.03%, while requiring a minimum ferrite levels (greater than 8%) to assure adequate resistance to IGSCC.

For any highly irradiated stainless steel in-core components such as control blades and instrument tubes, it is desirable to provide additional potential for the component to reach the intended design life. Consequently, for the ABWR, both the component design and materials of construction have been altered to provide this higher resistance against irradiation-assisted stress corrosion cracking for IASCC. The ABWR designs are crevice-free and incorporate control of trace element chemistry such as sulfur and phosphorus that provide this resistance.

The final material selected for specialized ABWR applications is XM-19. This is a high chromium, high manganese austenitic stainless steel alloy that is strengthened with specific nitrogen addition (as opposed to carbon). It has an established record of excellent performance in very demanding applications in BWR service where both high strength and extremely high resistance to IGSCC are required. Example applications in the ABWR include components in the FMCRD as well as core plate and top guide bolting.

58.4.2.2 Nickel Base Alloys for ABWR/ESBWR Application Nickel base alloys, particularly Alloy 600 and X-750, are also used extensively in BWRs. In addition to their excellent corrosion resistance, nickel alloys have a thermal expansion coefficient more similar to the low alloy steel used in the construction of the RPV and nozzles. This feature has led to the application of Alloy 600 for several parts of the core internals, including the shroud support structure as a transition material placed between the alloy steel reactor vessel and the stainless steel internals. The other desirable feature of Alloy 600 is that it is resistant to stress corrosion crack initiation in the postweld heat-treated condition, thereby allowing attachment to low alloy steel prior to stress relief. For the ABWR, to add margin against IGSCC, these alloys and their weld metals have been modified with stabilizing additions of niobium to reduce the potential for chromium depletion. For the wrought structures, the ABWR employs an Alloy 600 (designated as 600M) that has a niobium content on the order of 1-3%. For weld metal, Alloy 82 with high stabilizing ratios is used, leading to high IGSCC resistance as well.

Alloy X-750 is also used to a limited extent in the ABWR where high strength is required. Alloy X-750 is a precipitation hardening high strength nickel-base alloy that will perform well in the ABWR environment under the proper application. To improve SCC resistance for the ABWR, both the heat treatment and stresses of Alloy X-750 are controlled, based on the extensive operating plant experience and material understanding.

58.4.2.3 Component Fabrication and Design Considerations Aside from selection of materials that are intrinsically resistant to IGSCC, the fabrication of components will be controlled to ensure that this high resistance is maintained in the finished part. Low heat input welding processes are used to decrease the likelihood of any sensitization due to the welding process. The margin of IGSCC resistance will be reduced even for the intrinsically resistant materials by abusive fabrication practices such as excessive cold working. Postmanufacturing surface processing such as polishing can be applied to the weld heat-affected zones to remove surface cold work and residual stresses and strains, thereby further increasing the resistance to IGSCC initiation. Contamination with in-process materials having high levels of species such as chlorine, fluorine, and sulfur can degrade performance. Consequently, to protect the high level of resistance obtained by using materials such as described above for the ABWR, the entire fabrication and installation process is controlled to prevent detrimental practices.

A final consideration in control of IGSCC is the design itself. Crevices have been eliminated where possible and the number of welds has been reduced. The top grid structure, for example, is manufactured from a single solution annealed plate, thereby eliminating the risk of IGSCC. The core shroud is constructed in a manner to locate all welds away from the highest fluence locations.

58.4.2.4 Water Chemistry Control With respect to materials performance in the BWR, a substantial body of data now exists that shows that water chemistry is a key factor in material degradation as well as radiation buildup processes, especially for core internal components. It has become very clear that the presence of oxidizing species in the high purity coolant (such as oxygen and hydrogen peroxide), as well as anionic species that contribute to the coolant conductivity, can be correlated with incidence of IGSCC cracking as well as the rate of progression of any initiated cracks. Events such as resin intrusions can also reduce the resistance to SCC initiation. The benefits of operating with good water chemistry are very clear and the owners of operating BWRs (even plants that have not experienced severe degradation in water chemistry such as resin intrusions) have adopted practices to obtain low conductivity. Thus, the Electric Power Research Institute (EPRI) has developed guidelines for water chemistry control that provide specific recommended limits on overall conductivity as well as specific species. For the ABWR, recommended water chemistry is applied such that the units operate at or below a target conductivity of 0.1 μS/cm whenever the reactor system is greater than 200°F (93°C).

These guidelines also recommend the application of hydrogen water chemistry (HWC) when possible. The main function of HWC is to inject sufficient hydrogen into the condensate to reduce oxidizing species and maintain the electrochemical potential (ECP) of the reactor water below –0.23 V standard hydrogen electrode (SHE). The addition of hydrogen has been documented to effectively control SCC. Although HWC can result in additional main steam line radiation, the use of noble metal technology such as noble metal chemical addition (NobleChem™) can reduce the hydrogen needed for SCC control through catalytic action while limiting any steam line radiation issues.

58.4.2.5 Material Considerations for 60-Year Design Life The reactor vessel is also a very important component where 30 years of BWR operating experience and materials understanding have been used in the ABWR. The vessel draws from the understanding of material degradation mechanisms and fabrication experience. Of key interest are the effects of neutron irradiation over the life of a reactor vessel beltline region (the region immediately around the core), which results in a progressive loss of fracture toughness. This shift in NDT to higher temperatures can influence reactor operation with respect to such items as bolt-up temperature and hydrostatic test temperature. Consequently, it is advisable to start with a low NDT and use material resistant to neutron radiation damage. Copper and phosphorus have been identified as detrimental elements in alloys steels with respect to neutron damage, and nickel is considered to also be a factor. The corrosion behavior of reactor vessel steels is also well understood. There is significant experience with these steels to assure that there is high resistance to any SCC processes under the ABWR operating coolant environment.

For the ABWR reactor vessel, high toughness and retention of that toughness in service is provided. Strict controls are specified on initial fracture toughness of vessel components. Reactor vessel shell courses are required to have RT_{NDT} of –20°C or less. In addition, copper content is controlled to 0.05% maximum for base metal and weld metal. Phosphorus is limited to 0.006 and 0.008%, respectively. With these very low levels of copper, the shift in NDT for the ABWR vessels is expected to be very small over the operating life of the units. The estimated shift in RT_{NDT} is estimated to be less than 11°C over a 60-year operating life.

Low alloy steels are also used in many piping applications. The reduced corrosion rates make it attractive for several systems. However, the ABWR also employs carbon steel piping in some systems. This selection has been made to aid in plant fabrication while taking advantage of the steel's inherent toughness. Although erosion–corrosion in carbon steel power plant piping systems has been experienced in some units, oxygen injection is used for the ABWR to minimize this concern. Corrosion allowances for 60 years operation are included in the piping design.

58.4.3 Environmental Fatigue Rules

Since the early 1980s, the effects of high temperature water on the fatigue cyclic life of LWR components have been extensively discussed by numerous researchers. References [10]–[12] are some of the examples. The Subgroup on Fatigue Strength of the ASME Boiler & Pressure Vessel Code is currently working on a Code Case that would provide procedures for incorporating the reactor water environmental effects in the fatigue evaluation conducted as per the guidelines in Section III Paragraphs NB-3200 and NB-3600 [13].

Recently, the NRC has published Regulatory Guide 1.207 to provide guidance for determining the acceptable fatigue life of ASME pressure boundary components, with consideration of LWR environment [14]. The associated detailed guidance document is NUREG/CR-6909 [12]. NUREG/CR-6909 adopted the environmental fatigue correction factor method or F_{en} method to account for the environmental fatigue effects. F_{en} is defined as the ratio of fatigue initiation life in air at room temperature to that in reactor water at the service temperature. The regulatory guides are not mandatory. However, the NRC is likely to ask applicants for certification of new reactor designs for the technical approach they plan to follow to address environmental fatigue effects.

Reference [15] describes the results of the application of Regulatory Guide 1.207 methodology to an ABWR plant piping system. The system chosen was feedwater piping inside the containment. This system is typically classified as Class 1 as per the ASME Code classification.

Figure 58.13 schematically shows the feedwater piping system. The piping system delivers the feedwater to the reactor. It also receives water from RHR and RCIC systems. The portion of the piping between the reactor nozzle and the header at the containment penetration is designed as per ASME Class 1 requirements. Piping thickness is as per Schedule 80. The specified design pressure and temperature for this piping are 1250 psi and 550°F, respectively. The feedwater temperature during normal operation is 420°F.

Table 58.4 provides a summary of the calculated values of cumulative fatigue usage factors at two locations. For the feedwater nozzle location, usage factors are provided for both the nozzle side (low alloy steel) and the safe end side (carbon steel). It is seen that there is a modest impact on the calculated fatigue usage factors when reactor water environmental effects are factored in. At the safe end location, the reduction in air fatigue usage through the use of NUREG/CR-6909 S–N curves essentially offset the increase due to the use of F_{en}.

The increase in calculated fatigue usage when environmental fatigue effects are taken into account was modest for the feedwater piping considered in this evaluation. One of the reasons is that the normal operating temperature for the feedwater line (240°C) is comparatively lower than the typical operating temperatures for the primary piping in LWRs. In the case of carbon and low alloy steel piping systems, the increase due to the use of F_{en} is significantly offset by the advantage gained through the use of air S–N curves provided in NUREG/CR-6909. This would not be the case for stainless steel piping systems where the air S–N curves in NUREG/CR-6909 predict higher usage factor than the Code curve.

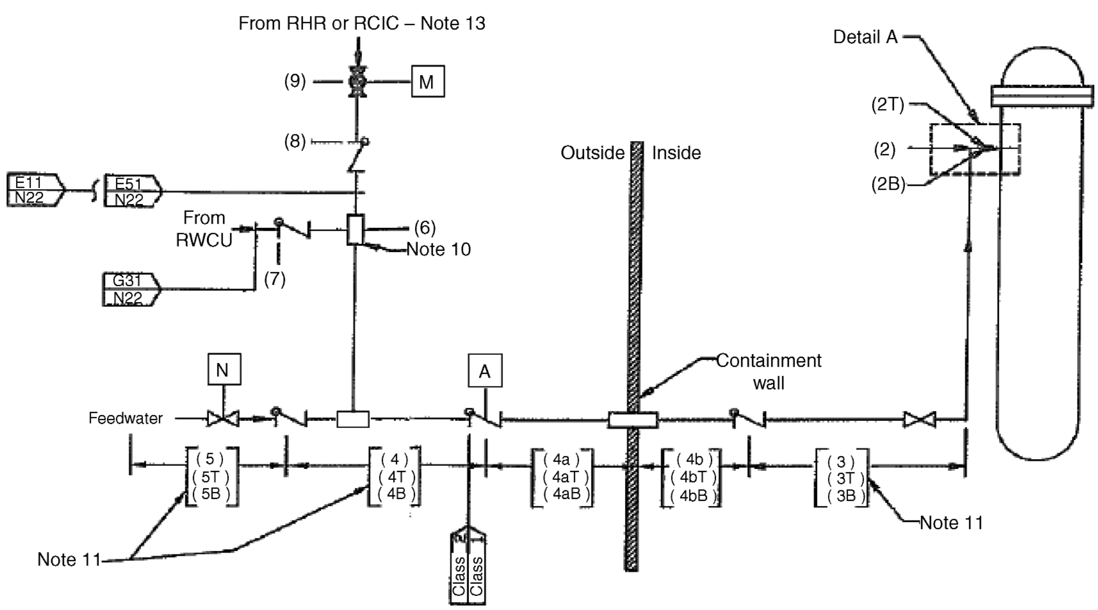

FIG. 58.13 SCHEMATIC OF AN ABWR FEEDWATER PIPING SYSTEM [15]

TABLE 58.4 CURRENT CODE AND ENVIRONMENTAL FATIGUE USAGE FACTORS FOR AN ABWR FEEDWATER LINE [15]

Node/location/material	Fatigue usage by current code	Fatigue usage by NUREG/CR-6909
Node 26/header/CS	0.083	0.117
Node 48/nozzle/LAS	0.085	0.302
Node 48/safe end/CS	0.085	0.086

In general, one would expect several fold increase in the calculated fatigue usage factor when F_{en} is used. This would have implications in terms of number of locations where hypothetical pipe breaks need to be postulated. Currently, the NRC Branch Technical Position MEB 3–1 [16] is used for postulation of breaks in high energy lines. MEB 3–1 requires postulation of a break at an intermediate locations if the fatigue usage at a location exceeds 0.1 or the primary plus secondary stress range exceeds 2.4 S_m. The calculated primary plus secondary stress range is not impacted by the use of F_{en} but the fatigue usage factor is. The use of F_{en} results in more locations where cumulative fatigue usage factor would exceed 0.1. More break locations means more pipe whip restraints to meet the requirements of General Design Criterion 4 of 10CFR50 [17]. However, the presence of more pipe whip restraints adversely affects the ability to conduct piping ISIs and thus have a negative impact on piping reliability during operation. The 0.1 fatigue usage threshold was based on engineering judgment and perhaps can be revised upward to say 0.4 or 0.6 to avoid this situation. The revision could be justified through a piping reliability analysis somewhat similar to that conducted in support of revised Appendix L in ASME Section XI Code [18].

58.4.4 Applicable ASME Code Edition

For ABWR, 1989 Edition of Section III of ASME Code was used for plant design and fabrication. The ESBWR uses two different Editions of Section III. The 2001 Edition including Addenda through 2003 Edition is used for all of the plant designs except for building construction. For building construction, 2004 Edition is used.

The ASME Section XI Edition committed to the ESBWR Design Certification Document (DCD) is 2001 Edition with Addenda through 2003. For licensing for specific site, the plant owner would select based on the permit date.

58.5 FUTURE DIRECTION – FABRICATION AND MODULARIZATION

Modularization techniques are implemented to reduce costs and improve construction schedules. These techniques are applied to such reactor fuel building items as (1) building reinforcing bar assemblies, (2) structural steel assemblies, (3) steel liners for the containment and associated water pools, (4) selected equipment assemblies, and (5) drywell platform and piping supports.

58.6 SUMMARY

The ESBWR represents an entirely new approach to the way nuclear plant projects are undertaken, modeled after the successful process used for ABWR. The ABWR was licensed and designed in detail even before construction began. Once construction did begin, it proceeded smoothly from start to finish in just 4 years.

Natural circulation is a proven technology that provides numerous benefits. Natural circulation allows the elimination of several systems, including recirculation pumps and associated piping, valves, heat exchangers, motors, adjustable speed drives, and controllers. The larger RPV employed for natural circulation provides synergy with the use of passive ECCS and improves the response to operational transients and increased safety margins. Flow transients resulting from recirculation pump anomalies are not present; that is, no runbacks or trips that would challenge stability.

The successful design, licensing, construction, and operation of the ESBWR nuclear power plant will usher in a new era of safe, economic, and environment-friendly nuclear electricity. The ESBWR is the first of a new generation of nuclear plants equipped with advanced technologies and features that raise plant safety to new levels, which significantly improve the economic competitiveness of this form of generation.

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