NATIONAL NUCLEAR LABORATORY

Boiling Water Reactor Technology - International Status and UK Experience

Position Paper

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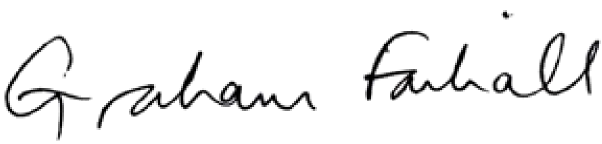
A position paper from the UK National Nuclear Laboratory

Following the recent acquisition of Horizon Nuclear Power by Hitachi-GE Nuclear Energy Ltd, there is the possibility that Boiling Water Reactors (BWRs) will be built in the UK over the next decade. The Generic Design Assessment (GDA) process has already begun, with the UK's Office for Nuclear Regulation (ONR) and Hitachi-GE working together, the result of which is intended to be authorisation for Hitachi-GE's Advanced BWR to be constructed in the UK. BWR technology is new to the UK, although some commonality exists between BWRs and the PWR technology which the UK has already deployed.

The purpose of this paper is therefore to give a description of BWR technology and to set out the basic design and operating principles of the BWR, with a more in-depth look into Hitachi-GE's ABWR . The evolution of the BWR is discussed, followed by a global overview of BWR deployment. The reactor core and nuclear fuel are then described, with particular emphasis on the differences and similarities between BWRs and PWRs (of which the UK has in-depth knowledge via Sizewell 'B' PWR).

A brief summary is made of the design features of the ABWR and, looking further into the future, the ESBWR. Whilst there are a number of important differences (discussed in this paper) between BWRs and PWRs much of the underpinning technology is similar and significant opportunities will exist for the UK supply chain to participate in the licensing, equipment manufacture, construction, commissioning and operation of BWRs.



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Introduction

Boiling Water Reactor (BWR) technology is well-established in several parts of the world, but has historically not been a significant feature of the UK nuclear industry. The recent announcement that the Horizon nuclear new build project has been taken over by Hitachi-GE, who are looking to develop BWRs on the Wylfa and Oldbury sites, has changed that.

This position paper summarises some of the key features of BWRs and their associated fuel cycles, the role BWRs play in the global industry and the experience in the UK – both in terms of operating such technology in Britain and of the UK experience which can help support a BWR programme.

Major design differences between PWR and BWR

BWRs are the second most common type of nuclear reactor worldwide accounting for about 20% of global installed nuclear generating capacity. The equivalent share for Pressurised Water Reactors (PWRs) is about 67%.

The fundamental difference between a BWR and a PWR is that water is allowed to boil in the BWR core and is then used directly to drive a turbine and generate electricity. The primary circuit pressure in a BWR is typically around half that necessary in a PWR (7 MPa compared with 15 MPa).

By using this 'direct cycle' strategy a BWR has no requirement for any of the secondary circuit components which are necessary in a PWR.

A PWR precludes boiling in the primary circuit by means of the high primary circuit pressure. The thermal energy from the primary circuit is used to raise steam in a secondary circuit via large steam generators – typically 4 per reactor in modern PWRs. This secondary circuit steam is then used to drive the PWR turbine generator to produce electricity for the grid.

Reactivity control also differs between the two designs. In a PWR the core reactivity is principally controlled by the use of boric acid (H_3BO_3) which is dissolved in the primary coolant. The concentration of the neutron absorbing boron is varied to maintain criticality. The control rods are

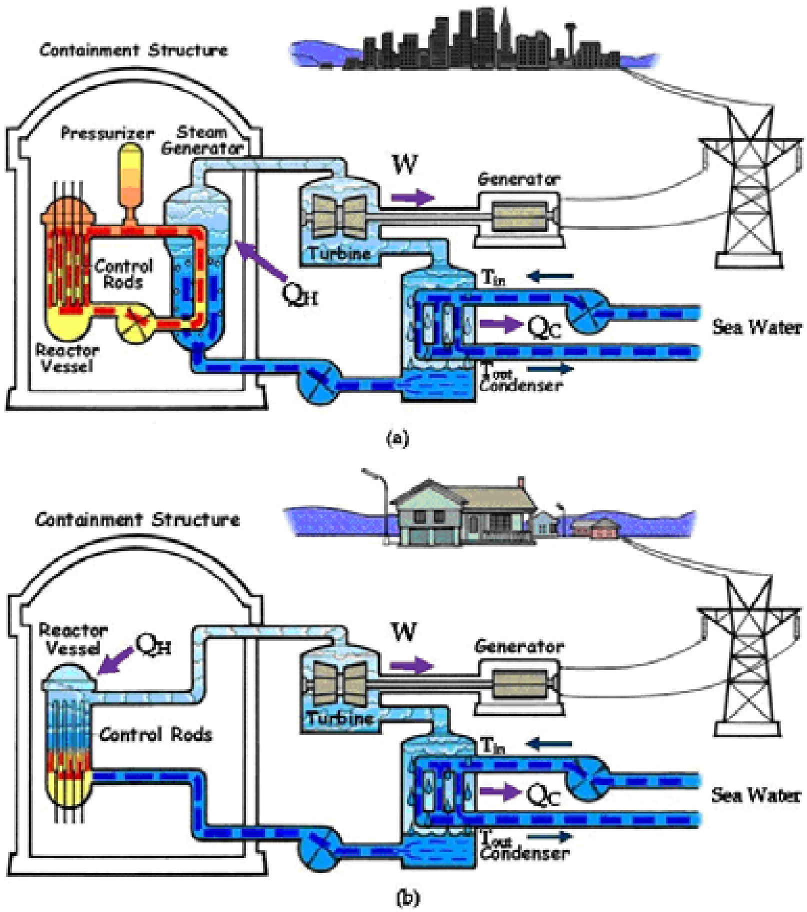


Fig 1 : Schematic comparison of PWR (top) and BWR (bottom) reactor designs (Source: US NRC) usually only shallowly inserted at the top of the PWR fuel assemblies during routine full power operation – the bulk of the reactivity control being achieved by the boron in the coolant.

Since the BWR has boiling water in the reactor core, boric acid cannot be used as the primary means of reactivity control. Steam bubbles will be increasingly prevalent as the coolant travels along its flow path up the reactor core. If boric acid were to be used, the formation of these bubbles would result in significant reactivity variations since the neutron absorbing boron would be largely absent from such bubbles (along with the majority of the boiling water). For this reason, reactivity management in a BWR is accomplished via (i) burnable poisons such as gadolinia (Gd_2O_3) intimately mixed with the UO_2 fuel, (ii) deeper control rod insertion and (iii) changing the amount of neutron moderation in the reactor via the control of coolant flow, eg in some BWR designs by varying the reactor coolant pump speed.

Since the water density decreases substantially with height in the BWR, the control rods (which are in fact shaped like cruciform blades) are inserted from the bottom of the core. The control rods would have less effect at the top of the core since the reduced neutron moderation (as a result of increased void formation) yields fewer thermal neutrons. The neutron absorbing material in the BWR control rods (boron in the form of B_4C) preferentially absorbs thermal neutrons therefore the control rods are most efficient in the low void regions towards the bottom of the reactor core.

Although a BWR has fewer components compared with a PWR (no steam generators, less piping) it should be noted that the BWR pressure vessel is much taller than for an equivalent PWR since steam separators and driers are located above the reactor core.

Also in a BWR, the primary circuit extends much further – including out to the turbine generator. This means that the same water which passes the fuel rods also passes through the turbine, and means that any water-borne contamination (for instance from a failed fuel rod) or activation products will be transported through the turbine and condenser. This requires additional shielding and access control of these components during reactor operation, although the restrictions are usually lifted once the reactor is shut down, since most activation products have very short half-lives (measured in seconds).

“Reactivity is controlled in PWRs primarily by varying the concentration of boric acid dissolved in the coolant. In BWRs, control rod movement and coolant pump speed variation are used.”

Evolution of BWR Technology

The first boiling water reactor experiments (BORAX-1) were carried out at the Argonne National Laboratory (ANL) in Idaho in 1953. These tests demonstrated that in-core boiling would be stable and that key operating characteristics (such as the void coefficient) were favourable from a safety standpoint. The first experimental boiling water reactor (EBWR) was then built by ANL in 1956 near Chicago. The development of commercial BWRs was undertaken by General Electric (GE) and the prototype Vallecitos BWR was built in 1957 near San Jose, California. This then led on to the development of commercial BWRs.

The first US commercial nuclear power plant, Dresden 1, was a BWR/1. The design then evolved through several variants of reactor and containment up to GE version BWR/6 as summarised in the table below. The majority of BWRs in operation worldwide are GE designed. ABB-Atom (now integrated with Toshiba-Westinghouse) and Siemens-KWU have also successfully built BWRs in Sweden and Germany respectively. In addition there are four Advanced Boiling Water Reactors (ABWRs) designed by Hitachi-GE and Toshiba in operation in Japan with a further two under construction.

The BWR containment design evolved in parallel with the reactor design, from the first BWR containments which were spherical "dry" structures. Dry containments are still used today in PWR designs. The BWR, however, switched to the "pressure suppression" containment design with a suppression pool. This had advantages including a higher heat capacity, an enhanced ability to accommodate rapid depressurisation and a simplified compact design. The Mark I containment was the first of the new containment designs. The Mark I design has a characteristic light bulb configuration for the reinforced concrete drywell, surrounded by a steel torus that houses a large water pressure suppression pool. This toroidal pressure suppression pool also plays an important role in the Mark I design as a heat sink in the case of an incident or accident.

The conical Mark II design has a less complicated arrangement. A key feature is the large containment drywell that provides more room for the steam and Emergency Core Cooling

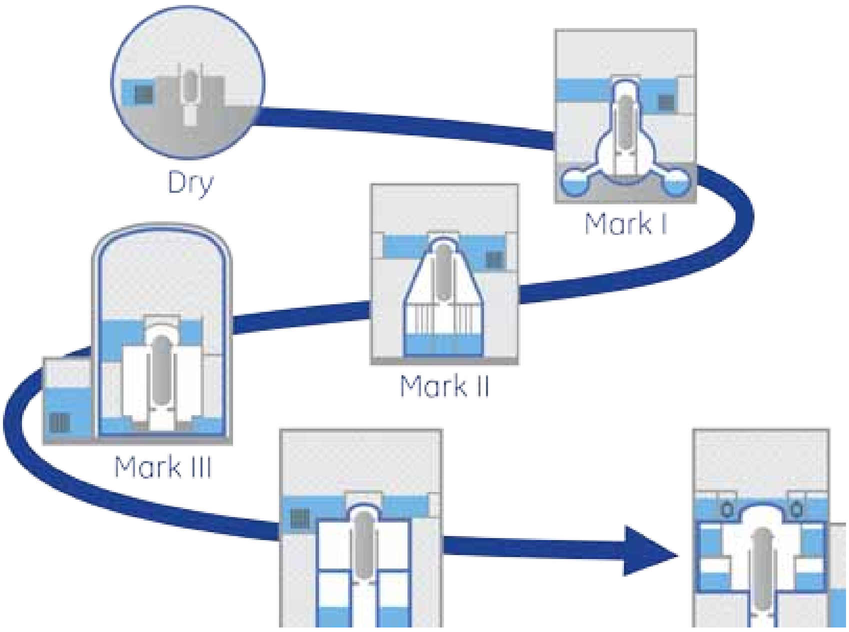


Figure 2: Evolution of GE BWR design (reproduced by permission of GE Hitachi)

Reactor	First Commercial Operation Date	Representative Plant/Characteristics
BWR/1	1960	Dresden 1 - Initial commercial size BWR
BWR/2	1969	Oyster Creek - Plants purchased solely on economics; large direct cycle
BWR/3	1971	Dresden 2 - First 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 protection system
ABWR	1996	Kashiwazaki-Kariwa 6 - Reactor internal pumps; fine motion control rod drives; advanced control room; digital and fibre optic technology; improved ECCS; high/low pressure flooders
ESBWR	Not yet applicable	Natural circulation; passive ECCS

ECCS - Emergency Core Cooling System

Figure 3: BWR Design Evolution

System (ECCS) piping. The Mark III containment design, used worldwide with BWR/6s and some BWR/5s, represented a major improvement in simplicity. Its containment structure is a right circular cylinder that is easy to construct, and provides ready access to equipment and ample space for maintenance activities. Other features of the Mark III include horizontal vents to reduce overall loss-of-coolant accident (LOCA) dynamic loads and a freestanding steel structure to ensure leak tightness. In Mark II and later BWR designs, the heat sink is provided by a 'wetwell', which is no longer in the form of a distinct torus but is integrated within the primary containment below the 'drywell'.

During faults, heat dumped into the wetwell via steam is transferred to the ultimate heat sink (eg, a river or a cooling tower) via heat exchangers.

In the ESBWR, the design is fully passive, meaning that no power systems or human interventions are needed for at least 72 hours to ensure the heat can be dispersed.

BWR Global Overview

There are 84 commercial BWRs in operation worldwide, making the BWR the second most common type of reactor behind the PWR (of which there are 272). The majority of the BWRs are sited in the US (35 reactors) and Japan (26 reactors). In addition, Sweden has mostly adopted BWR technology and BWRs are also operating in Finland, Spain, Switzerland, Germany, Mexico and India.

The BWR is a mature robust reactor design with a performance record comparable with that of the PWR. As regards operational experience, BWRs and PWRs are the top performing reactors with modern variants of both designs performing equally well in terms of availability.

Fukushima

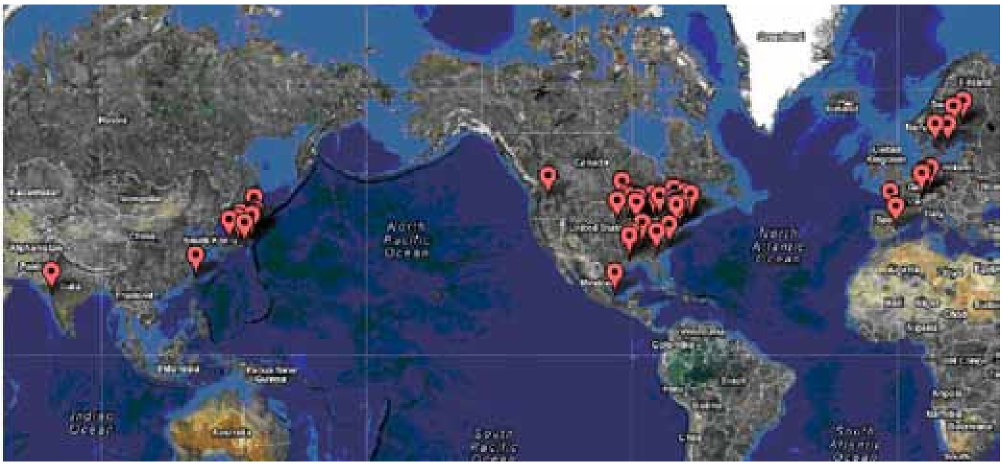
The Fukushima incident in March 2011 was centred around the response of a number of BWR reactors to a major earthquake and associated tsunami. The Fukushima site was home to 6 GE-designed BWRs, which began operating in the 1970s.

Fukushima Unit 1 is a BWR/3 design (see table above), whereas Units 2-5 are BWR/4 technology and Unit 6 is a BWR/5 design.

Country	PWR		BWR	
	No.	MW(e)	No.	MW(e)
ARGENTINA				
ARMENIA	1	375		
BELGIUM	7	5927		
BRAZIL	2	1884		
BULGARIA	2	1906		
CANADA				
CHINA	13	10496		
CZECH REP.	6	3766		
FINLAND	2	976	2	1760
FRANCE	58	63130		
GERMANY	7	9496	2	2572
HUNGARY	4	1889		
INDIA			2	300
IRAN, ISL. REP.	1	915		
JAPAN	24	19284	26	24931
KOREA, REP. OF	17	15966		
MEXICO			2	1300
NETHERLANDS	1	482		
PAKISTAN	2	600		
ROMANIA				
RUSSIA	17	12864		
SLOVAKIA	4	1816		
SLOVENIA	1	688		
SOUTH AFRICA	2	1830		
SPAIN	6	6057	2	1510
SWEDEN	3	2811	7	6515
SWITZERLAND	3	1700	2	1563
UK	1	1191		
UKRAINE	15	13107		
USA	69	67368	35	34097
TOTAL	270	248364	84	77726

Figure 4: List of operating LWRs and Map showing LWRs around the globe (Source: WNA)

The circumstances relating to the events of March 2011 are recorded in detail elsewhere and this paper does not attempt to summarise those events here. However BWR technology has advanced substantially between the reactors which were damaged at Fukushima and today's modern designs such as ABWR and ESBWR. As outlined earlier, these improvements include containment design, reduced fuel duty (power per metre of fuel rod), improvements to the emergency core cooling systems and – in the case of ESBWR – a move to passive safety systems.



Fuel Differences between PWR and BWR

In principle, BWR fuel is similar to that used in PWRs -UO_2 fuel pellets are stacked within sealed pins clad in zirconium alloy. These pins are then restrained in rigid square arrays to make fuel assemblies.

There are however some fundamental differences between PWR and BWR fuel, in particular in relation to the design of the fuel assemblies.

- While the active fuel heights are broadly comparable for the two systems (approximately 3.5 m) the BWR fuel assembly is ~¼ the size of the PWR variant. The PWR fuel is typically a 17x17 array of fuel rods whereas the BWR typically comprises a 10x10 array (or smaller). This is to accommodate the BWR cruciform control rods which move in the gaps between every group of 4 fuel assemblies to help control reactivity. As explained earlier, reactivity control in BWRs cannot be achieved through the use of boric acid in the coolant, since the water coolant is designed to boil, producing significant voidage, which would reduce the effectiveness of this approach

- As well as being smaller, BWR fuel assemblies have a high degree of heterogeneity compared with PWRs. PWR fuel usually comprises a single ^{235}U enrichment radially, with perhaps a lower enrichment region at the top and bottom (i.e., axial blankets) to improve neutron economy

- BWRs however have 4 or 5 different enrichments radially within a single assembly for power distribution optimisation (as well as doped pins incorporating a neutron poison for reactivity control through the life of the fuel, such as

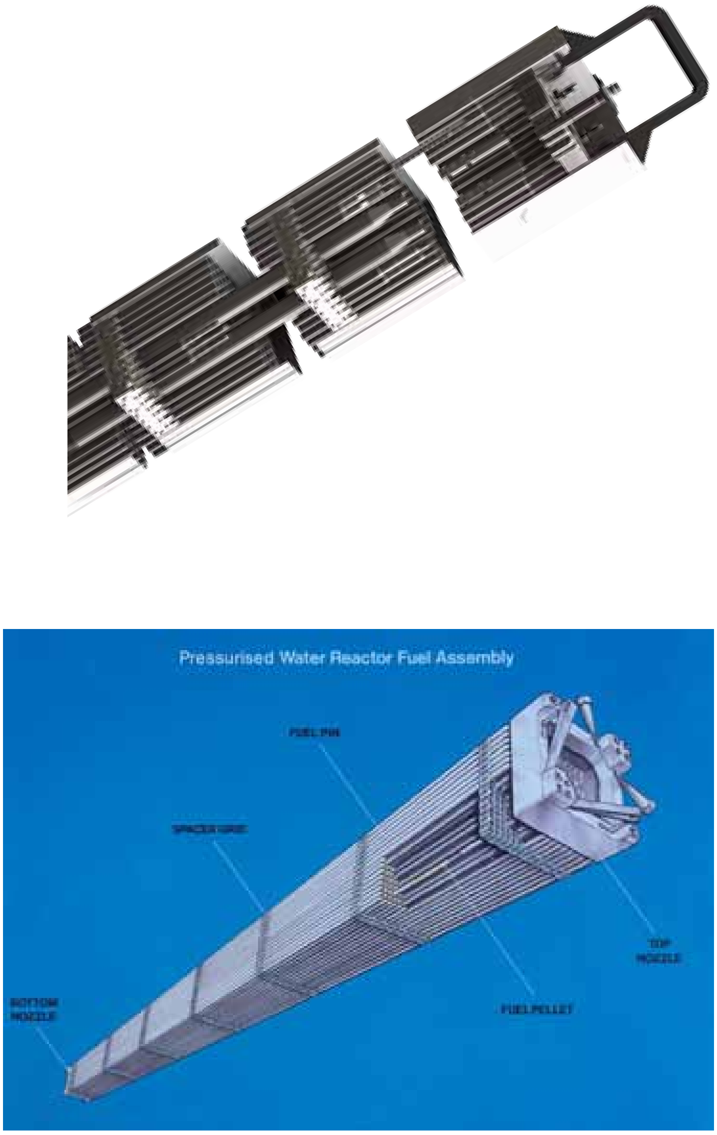


Figure 5: Schematic of 4 BWR fuel assemblies plus control rods (top, reproduced by permission of GE Hitachi) and a PWR fuel assembly and control rod (reproduced by permission of Westinghouse).

Fuel Differences between PWR and BWR cont.

Gd_2O_3 or Er_2O_3). The fuel enrichment also varies axially.

- In addition, part length fuel rods are often used in BWRs, water channels are included to improve moderation and the entire array of rods is encased in a zirconium alloy box to reduce cross-flow between assemblies, which could otherwise be encouraged by the combination of part-length rods, boiling coolant and control blade movement. In turn, cross flow could lead to localised "dry spots" on the clad wall and the associated risk to cladding integrity.

Fuel Cycle Implications

Mining

The fuel discharge burnups and throughputs for a BWR are similar to those seen in a PWR and so the mining requirements in terms of TWhe per kg of mined yellowcake (U_3O_8) are similar.

Enrichment

Again, the fuel enrichment requirements are broadly equivalent between the BWR and PWR although the BWR fuel requires a larger number of fuel enrichments within each assembly.

Conversion and Fuel Manufacture

Similar again for both BWR and PWR in terms of conversion volume requirements and intermediates going from U_3O_8 to enriched UO_2. The fuel manufacturing route is of course different given the more heterogeneous nature of the BWR fuel, the inclusion of a fuel shroud for each BWR assembly and the inclusion of part length rods and water channels.

Irradiation

BWRs can operate with enriched non-irradiated or reprocessed uranium fuel and Mixed plutonium/uranium OXide (MOX) fuels in the same manner as PWRs.

Reprocessing

If a closed fuel cycle is adopted, BWR fuel is fully capable of being reprocessed, in the same way as PWR fuel. There is extensive global experience of both reactor types operating in closed fuel cycles.

Disposal

If a reprocessing route is followed, the optimal long term disposal route for the fuel waste streams is fit for purpose for both PWR and BWR fuel (i.e. cementation of ILW, vitrification of HLW followed by geological disposal in a repository).
If an open fuel cycle is adopted, then BWR fuel can be stored in the same way as PWR fuel. Similarly, direct disposal of BWR fuel is comparable to that of PWR fuel.

"BWR fuel assemblies are more heterogeneous than PWR fuel, with several different fuel rod types within a single assembly (cf. one or two rod types in standard PWR fuel)."

Specifics of the ABWR Design

The Advanced Boiling Water Reactor (ABWR) was developed by General Electric, Hitachi and Toshiba during the 1980s under the sponsorship of Tokyo Electric Power Company (TEPCO). The design objective was to produce a reactor that combined the best features of existing BWRs with new technologies and modular construction techniques. Safety improvements were given the highest priority.

Figure 7 shows a comparison of ABWR with the BWR/6 design. It is worth noting that the last BWR/6 to come into operation was Hamaoka Unit 4 in Japan, which received its operating licence in 1993.

Compared with previous BWR designs the ABWR is reported to be safer, more reliable and easier to operate and maintain. The capital costs

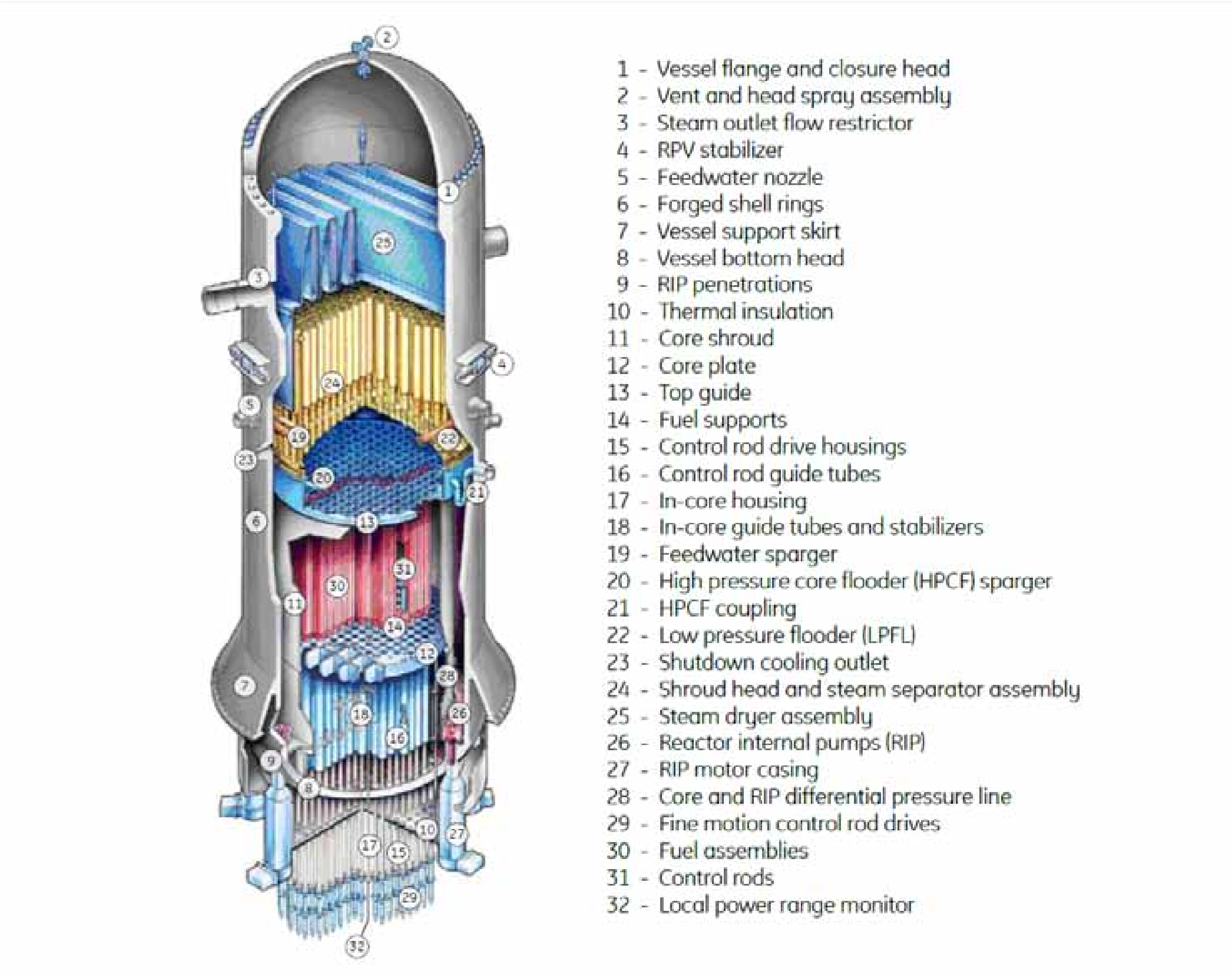


Figure 6: ABWR Schematic (reproduced by permission of GE Hitachi)

Specifics of the ABWR Design cont.

and the operation and maintenance costs are lower and it has a shorter construction time (eg, approximately 39 months from first concrete to first fuel load in Japan).

The internal coolant pumps used in the ABWR eliminate external recirculation systems (so there is less external pipework and therefore less risk of a LOCA). The more compact integrated containment and reactor building has an improved seismic response and is easier to construct. Efforts were also made to reduce radwaste and occupational exposure, for example by minimising the use of stellite (a cobalt-chromium alloy). Stellite is a very wear-resistant alloy typically used in valve seats (its properties also make it suitable to line machine gun barrels). However, the ^{59}Co in stellite will absorb neutrons to become ^{60}Co which decays with a five year half-life releasing beta radiation and very energetic gamma rays. The minimisation of its use therefore reduces occupational exposure.

The ABWR is fully automated in response to a loss of coolant accident (LOCA), and operator action is not required for 3 days. After 3 days the operators must replenish ECCS water supplies. These and other improvements make the plant safer than previous reactors.

Feature	AWBR	BWR/6
Recirculation	Vessel mounted reactor internal pumps	Two external loop recirculation systems with jet pumps inside RPV
Control Rod Drives (CRD)	Fine motion CRDs	Locking piston CRDs
ECCS	3-division ECCS	2-division ECCS plus HPCS
Reactor vessel	Extensive use of forged rings	Welded plate
Primary containment	Advanced - compact, inerted with nitrogen	Mark III
Secondary containment	Reactor building	Shield, fuel, auxiliary and DG buildings
Control and instrumentation	Digital, multiplexed, fibre optics, multiple channel	Analog, hardwired, single channel
Control room	Operator task-based	System-based
Severe accident mitigation	Inerting, drywell flooding, containment venting	Not specifically addressed
Offgas	Passive offgas with room temperature charcoal	Active offgas with chilled charcoal filters

CRD Control Rod Drive
ECCS Emergency Core Cooling System
HPCS High Pressure Core Spray
DG Diesel Generator

Figure 7: Comparison of key ABWR and BWR/6 features

ABWR construction and operating experience

There are four 1.3 GWe ABWR units in Japan; Kashiwazaki Kariwa 6 & 7, Hamaoka 5 and Shika 2. A further 2 units are under construction at Matsue and Oma. There are also 2 ABWRs being built at Lungmen in Taiwan.

The ABWRs at Kashiwazaki-Kariwa (the world's highest output nuclear power plant complex) were built in 36.5 and 38.3 months for Unit 6 and 7 respectively, approximately 10 months less than the previous quickest construction time for a BWR in Japan. Despite being first of a kind they were also built to budget.

From the IAEA Power Reactor Information System the average load factor for these plants is of the order of ~45%.

The reasons for this low value for the ABWR are:

- ABWRs have only been operating for a relatively short period of time (since 1996, 1997, 2005, 2006) therefore a 'poor year' skews the load factor data disproportionately compared with established plants that have been operating for longer;
- ABWR outages have been due to earthquakes (in 2007 at Chūetsu and in 2011 at Fukushima) and turbine blade problems;
- Damaged turbines were found at Hamaoka 5 and Shika 2 reactors. At the Kashiwazaki-Kariwa 6 and 7 plants, damage was also found in turbines though to an extent that did not adversely affect operation. All damage was due to the design of the 52-inch turbine blades. The design was subsequently changed to make the turbine blades more resistant to both random and 'flashback' vibrations. Random vibrations are irregular oscillations that occur in the blades as a result of steam turbulence within the turbine. They occur when the turbine is under no load or at low load operation. Flashback vibrations are caused by wet steam that enters the turbine during a load rejection. This reverse flow of steam into the turbine is caused by the pressure drop in the turbine that occurs when the steam flow is stopped or significantly reduced;
- The load factors of the Kashiwazaki-Kariwa 6 and 7 plants before the 2007 earthquake - over 80% - are more indicative of the real ability of the ABWRs, and are equivalent to those of Japanese BWRs (62%) and PWRs (78%) pre-Fukushima;
- At Hamaoka 5, a shutdown was caused by a high hydrogen concentration in the off-gas system. The off-gas system processes and filters the hydrogen, oxygen and noble gases (eg, krypton and iodine) from the BWR's condenser. The condenser cools the steam exiting the turbines before it is sent back as water through the core once more.
- The hydrogen concentration increased due to a decrease in the performance of the catalyst for recombining the hydrogen and oxygen in the system. This was due to two factors, the manufacturing process of the catalyst and a substance contained in the sealing material used for the turbine system. The same incident took place in other BWRs eg, Onagawa 3 and Hamaoka 4. Countermeasures were developed and duly adopted. (http://www.neimagazine.com/features/featureinside-japanese-outages-721/)
- To summarise, the unplanned shutdowns have been due to conventional problems and do not suggest there are any inherent ABWR-specific problems now that the turbine issues have been resolved. It can be anticipated that ABWR availability in future will be comparable to other modern PWR and BWR designs, as described earlier.

The Economic Simplified Boiling Water Reactor (ESBWR)

GE-Hitachi also offers the Economic Simplified Boiling Water Reactor (ESBWR) as part of their reactor portfolio. The ESBWR incorporates passive safety features derived from its predecessor, the Simplified Boiling Water Reactor (SBWR) and from the Advanced Boiling Water Reactor (ABWR).
The SBWR program was not pursued due to the economics of small reactor vs larger variants. However, aspects of the passive safety technology developed and tested during the SBWR program have been transferred directly into the ESBWR.

One of the main differences between the ESBWR and current operating reactors is the absence of reactor coolant pumps and no dependence on diesel generators for safety systems. The reactor operates using natural circulation, enhanced by using a taller reactor pressure vessel (RPV). The taller RPV also contributes to enhanced safety margins with a robust passive Emergency Core Cooling System (ECCS).

It is worth noting that natural circulation in BWRs is a proven technology, eg, it was used in some of the early plants like Dodewaard (183 MWth) and Humboldt Bay (165 MWth). The move was made to use forced circulation (using pumps) in order to achieve higher powers in a compact pressure vessel. Pressure vessel fabrication capability at the time was a factor in this decision.
The ESBWR Safety Systems design incorporates four redundant and independent divisions of the passive Gravity Driven Core Cooling System, the Automatic Depressurization System (ADS) and a Passive Containment Cooling System (PCCS) .

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, with no requirement for pumps. The water source needed for the ECCS function is stored in the containment upper drywell, with sufficient water to ensure core coverage to 1 meter 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 hours following a postulated design basis accident, and provisions exist for external makeup beyond that, if necessary.

The ESBWR draws upon proven ABWR technology and design. For example, it uses the same diameter reactor pressure vessel as the ABWR and some of the same internals. The ESBWR core was also increased in size by adding fuel assemblies to increase power level. Fuel height was decreased to 3.0 meters in order to achieve the appropriate pressure drop. The core was increased from the 732 fuel assemblies in the SBWR design to 1132 fuel assemblies in the ESBWR, resulting in a thermal power rating of 4500 MWth.

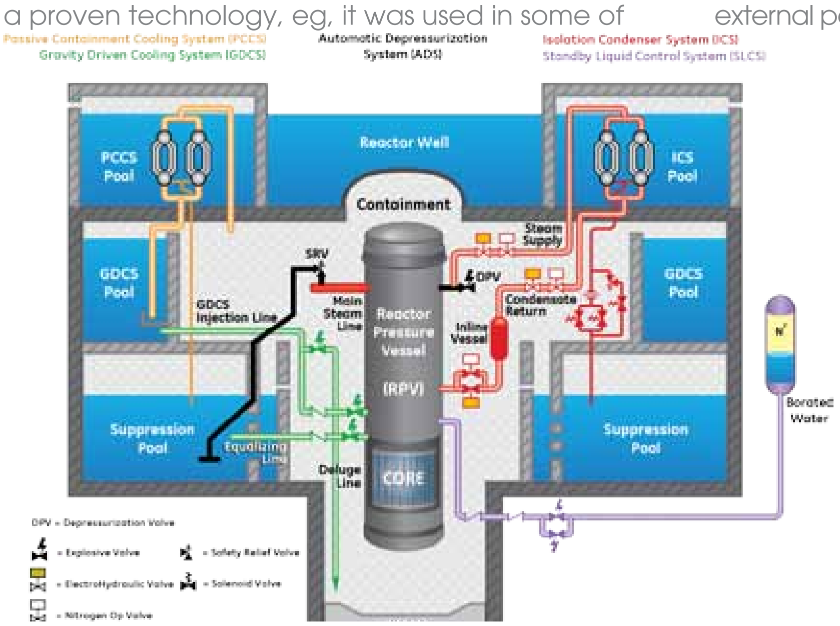


Figure 8: ESBWR Key Safety Systems (reproduced by permission of GE Hitachi)

Overview of BWR operating experience in the UK

The UK's past experience with operating BWRs is limited mainly to the UK-developed Steam Generating Heavy Water Reactor (SGHWR). The SGHWR operated from 1967 to 1990 at the UKAEA Winfrith site, 12 miles east of Dorchester. It was a prototypic reactor which shared some features of both the CANDU and BWR present day designs. The moderator was heavy water at atmospheric pressure physically separated from the primary light water coolant in the same manner as a CANDU (note however that CANDU reactors currently use heavy water for both moderation and cooling). A bank of pressure tubes passed through channels in the moderator vessel. Each pressure tube contained a low-enriched UO_2 fuel element comprising a bundle of 36 individual zirconium alloy clad fuel pins, each containing a stack of fuel pellets.

The light water within the pressure tubes was allowed to boil, raising steam which was sent directly to a turbine in a similar manner to modern day BWRs. The SGHWR operated successfully with a power output of 100 $MW_{electrical}$ (292 $MW_{thermal}$) for 23 years before being shutdown and decommissioned.

British Nuclear Fuels plc (BNFL) manufactured the fuel for the SGHWR at its Springfields site in Preston, Lancashire (now owned by Toshiba-Westinghouse). Springfields also manufactured fuel for the Dodewaard BWR in The Netherlands.

The UK has extensive experience of reprocessing both PWR and BWR fuel from overseas, and the National Nuclear Laboratory (NNL) has undertaken campaigns of Post Irradiation Examination (PIE) on BWR fuel from overseas, in order to evaluate its performance at high burnup levels.

Implications for UK Licensing

The Generic Design Assessment (GDA) process which has taken forward the licensing of two PWR designs (AP1000 and EPR) to Interim Approvals over recent years originally included two other designs – one of them a BWR. The GE-designed ESBWR was one of the four reactors included in GDA from its inception in Summer 2007 to the end of Step 2 in Spring 2008. At this point GE-Hitachi suspended their application, and work ceased. If ESBWR were to be taken forward again in the UK, that GDA application could be re-activated.

Obtaining licensing approval in the UK does not necessarily have to be done via the GDA process. It would be possible for a developer of a modern BWR design to proceed via a parallel site licence and reactor assessment processes, although it would be expected that much of the good practice from GDA (such as open communication, public consultation and regular reporting of metrics) would be retained. The overall requirements to demonstrate the safety of the reactor design are the same, irrespective of the process taken.

By doing a limited amount of work on the ESBWR, ONR and EA have accumulated some recent expertise in the assessment of BWR technology against UK safety and environmental requirements. It is expected that this would be useful in a future BWR assessment.

Figure 9: The SGHWR at Winfrith (reproduced by permission of NDA)

Overview of BWR operating experience in the UK cont.

Implications for UK supply chain

The fact that BWRs may feature in the first tranche of UK new build reactors, alongside PWRs, should not affect the UK unduly since there is a high degree of commonality between the two systems (once the absence of a secondary circuit is acknowledged). Electrical systems, pumps, pipework and much of the infrastructure associated with modern BWRs should be available for sourcing within the UK if required.

One potential issue could be that despite the UK having manufacturing capability for UO_2 fuel for the indigenous UK Advanced Gas Reactors and for PWR fuel at Westinghouse's Springfields Fuel Production Facility, fuel production lines specifically tailored to modern BWR requirements do not exist as yet in the UK.

Apart from this fact, the supply chain situation remains largely the same whether a PWR or a BWR is considered.

UK expertise and knowledge relevant to BWR technology

The majority of the UK's light water reactor expertise was developed in support of Sizewell 'B' PWR and encompasses the full fuel cycle from uranium procurement and PWR fuel manufacture to irradiation experience in Sizewell 'B' PWR. The UK also has the capability to carry out all the necessary steady state nuclear design and transient analysis (fault studies) for PWRs. While some of this methodology remains relevant to BWR design and performance, the UK would need to develop the capability to carry out the equivalent analyses for a BWR programme.

The UK also currently has the ability to reprocess oxide fuel (from either PWR or BWR reactors) at the Sellafield facility, although the Thorp plant is scheduled to close in 2018, which precedes any credible operational date for a new UK BWR. It should also be noted that current UK policy for new nuclear build is for an open fuel cycle, rather than a reprocessing route.

Specific BWR expertise is necessarily more limited, however as regards fuel manufacture. The Springfields fuel manufacturing facility produced the fuel for the UK's SGHWR and also the finished fuel for the Dodewaard BWR for approximately 20 years up until the 1990s. The Dodewaard fuel incorporated many of the design features seen in modern day BWRs such as fuel doped with neutron poisons and radial and axial enrichment zoning. In addition to the documented information which is relevant to BWR fuel design, a number of the key individuals who worked on this project remain within the industry and their expertise could be harnessed to develop a more extensive capability to support a UK BWR deployment.

The UK currently has some relevant experience in the field of BWR coolant chemistry, although again this resides in a limited number of experts and would need to be boosted to allow a bigger contribution to the UK new build programme of BWRs.

In addition, extensive generic nuclear skills exist such as criticality and shielding capabilities, environmental impact teams, and materials testing and analyses facilities. Finally, the skills, hardware, software and techniques used to date to carry out fuel manufacture, reactor safety, core design and licensing calculations for PWRs all place the UK nuclear workforce in a well-primed state to map these skills to BWRs via a transitional period of investment and training.

Conclusions

Boiling Water Reactor technology looks set to become a feature of the UK's fleet of nuclear plants, when the Horizon project comes on-line around the end of this decade.

Whilst there are a number of important differences between BWRs and the PWRs with which the UK is familiar – in terms of reactor design, fuel and so on - much of the underpinning technology is similar and there remain substantial opportunities for the UK supply chain to participate in the licensing, equipment manufacture, construction, commissioning and operation of BWRs.

Where gaps exist, there is, in many cases, scope for UK industry to take steps now to boost the depth and breadth of UK capability to increase the scope for involvement.

"The ABWR is the world's first "Generation III" nuclear reactor design to be built and operated. Substantial opportunities will exist for the UK supply chain to work in partnership with the reactor technology provider and the operating company."

References

Where images have been sourced separately, reference to the original source can be found below:

Figure 1

Source: US NRC

Figure 2

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Figure 3

http://www.ne.doe.gov./np2010/pdfs/esbwrGenera%20DescriptionR4.pdf

Figure 4

Map - WNA Nuclear Database

Table - http://www-pub.iaea.org/MTCD/Publications/PDF/RDS2-32_web.pdf

Figure 5

Top - Reproduced by permission of GE Hitachi
Bottom - Reproduced by permission of Westinghouse

Figure 6

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Figure 7

Economic Simplified Boiling Water Reactor Plant General Description June 2006 - http://www.ne.doe.gov/np2010/pdfs/esbwrGenera%20DescriptionR4.pdf

Figure 8

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