

**Status Report – BWRX-300 (GE Hitachi and Hitachi GE Nuclear Energy)  
USA**

**DATE (2019/9/30)**

The BWR-300 is the 10<sup>th</sup> generation Boiling Water Reactor (BWR) created by GE Hitachi Nuclear Energy (GEH). It is a SMR evolution of the ESBWR which is licensed by the US NRC and utilizes many of the components for the operational ABWR. The first BWRX-300s are expected to start construction in 2024 and 2025 and enter commercial operation in 2027 and 2028.

**INTRODUCTION**

Indicate which booklet(s): [ ] Large WCR [x] SMR [ ] FR

GE Hitachi Nuclear Energy’s (GEH’s) BWRX-300 is a designed-to-cost, 300 MWe water-cooled, natural circulation Small Modular Reactor (SMR) utilizing simple, natural phenomena driven safety systems. It is the tenth generation of the Boiling Water Reactor (BWR) and represents the simplest, yet most innovative BWR design since GE, GEH’s predecessor in the nuclear business, began developing nuclear reactors in 1955. The BWRX-300 is an evolution of the U.S. NRC-licensed, 1,520 MWe ESBWR. It is designed to provide clean, flexible energy generation that is cost competitive with natural gas fired plants. Target applications include base load electricity generation, load following electrical generation within a range of 50 to 100% power and district heating. GEH, a world-leading provider of advanced reactor technology and nuclear services, is a global alliance created by the General Electric Company (GE) and Hitachi, Ltd. to serve the global nuclear industry.

**Development Milestones**

- 2014 ESBWR DCD Issued
- 2017 BWRX-300 Evolution from ESBWR Initiated
- 2018 Pre-licensing engagement with the UK ONR
- 2019 Start of US NRC pre-licensing engagement including Licensing Topical Report Submittal
- 2020 Start of Canadian CNSC Vendor Design Review Combined Phase 1 and 2
- 2022 Submittal of construction permit application in US and Canada 2024/5 Start of construction in US and Canada
- 2027/28 Commercial Operation in US and Canada

Design organization or vendor company (e-mail contact): d.mcdonald@ge.com

**GEH Home Page:** <https://nuclear.gepower.com/build-a-plant/products/nuclear-power-plants-overview/bwrx-300>

The BWRX-300 is a designed-to-cost, 300 MWe water-cooled, natural circulation Small Modular Reactor (SMR) utilizing simple, natural phenomena driven safety systems. It is

being developed by GE-Hitachi Nuclear Energy (GEH) in the USA and Hitachi GE Nuclear Energy (HGNE) in Japan. It is the tenth generation of the Boiling Water Reactor (BWR) and represents the simplest, yet most innovative BWR design since the General Electric Company (GE), GEH's predecessor in the nuclear business, began developing nuclear reactors in 1955. The BWRX-300 is an evolution of the U.S. NRC-licensed, 1,520 MWe ESBWR. It is designed to provide clean, flexible energy generation that is cost competitive with natural gas fired plants. Target applications include base load electricity generation, load following electrical generation within a range of 50 to 100% power and district heating.

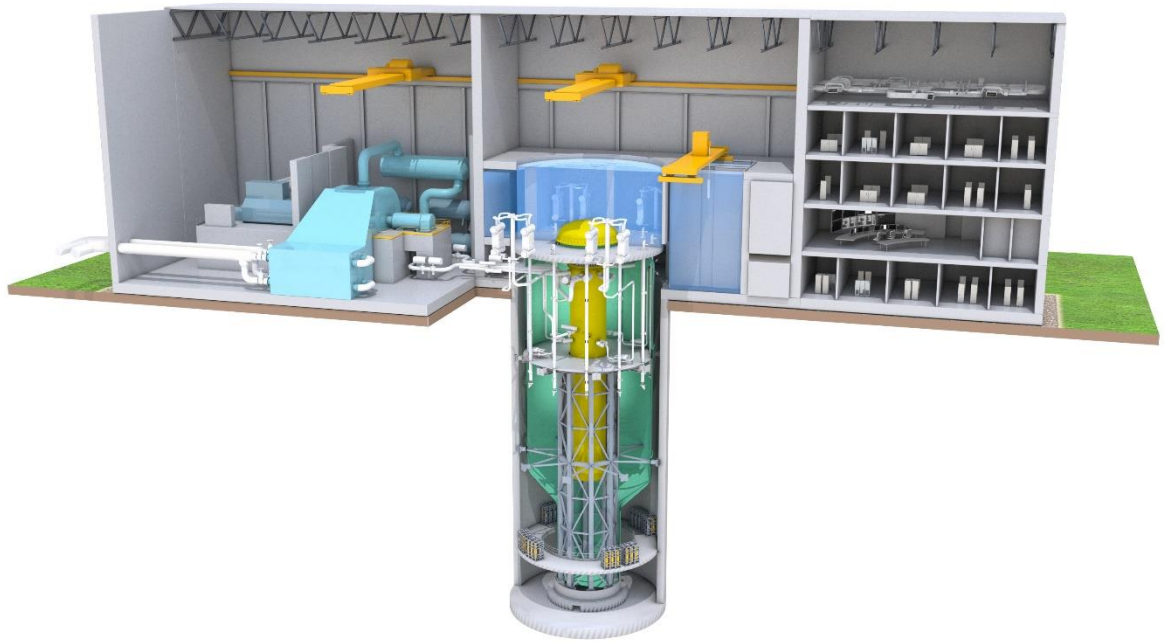
The BWRX-300 design optimizes the cost of construction, operation, maintenance, staffing and decommissioning. Cost are minimized while maintaining world class safety by implementing a safety assessment framework structured on the five defense lines of IAEA's Defense-in-Depth methodology. GEH's intense focus on minimizing all aspects of cost was driven by discussions with multiple GEH customers who indicated that new nuclear will only be built in significant quantities if it is cost competitive with all forms of new energy generation.

Focusing on cost was also borne out as the only path to significant new nuclear generation in a 2018 [EPRI report](#) that determined that advanced nuclear power generation is insensitive to displacing other power generating technologies in the USA by 2050 unless the capital cost is less than \$3,000 USD/kW. Additionally, [MIT research](#) determined that future nuclear power installations will need to come from proven nuclear steam supply system (NSSS) supply chains and standard, modularized, off-the-shelf equipment at an overnight EPC capital cost less than \$2,500 USD/kW.

The top-level features of the BWRX-300 include:

- 10th generation BWR technology
- 300 MWe SMR
- Comply with the international high safety standards
- Designed to be cost competitive with gas
- Up to 60% capital cost reduction per MW
- Evolved from the licensed ESBWR
- Designed to mitigate LOCAs with a simple, dry containment without relying on active safety systems Reduced on-site staff and security
- Design-to-cost approach: <\$1B USD first of a kind (FOAK) and <\$2,250 USD/kW nth of a kind (NOAK)
- Proven components, fuel, and supply chain
- Constructability integrated into design

Additional details can be found on the [GE Hitachi web site](#).



**Table 1: ARIS Category Fields (see also Spreadsheet “Categories”) for Booklet**

ARIS Category	Input	Select from
Current/Intended Purpose	Commercial – Electric, District Heating	Commercial – Electric/Non-electric, Prototype/FOAK, Demonstration, Experimental
Main Intended Application (once commercial)	Baseload	Baseload, Dispatchable, Off-grid/Remote, Mobile/Propulsion, Non-electric (specify)
Reference Location	Below-Ground	On Coast, Inland, Below-Ground, Floating-Fixed, Marine-Mobile, Submerged-Fixed (Other-specify)
Reference Site Design (reactor units per site)	Single Unit	Single Unit, Dual Unit, Multiple Unit (# units)
Reactor Core Size (1 core)	Small	Small (<1000 MWth),
Reactor Type	BWR	Medium (1000-3000 MWth),
Core Coolant	H <sub>2</sub> O	Large (>3000 MWth)
Neutron Moderator	H <sub>2</sub> O	PWR, BWR, HWR, SCWR, GCR, GFR, SFR, LFR, MSR, ADS
NSSS Layout	Direct-cycle	H <sub>2</sub> O, D <sub>2</sub> O, He, CO <sub>2</sub> , Na, Pb, PbBi, Molten Salts, (Other-specify)
Primary Circulation	Natural	H <sub>2</sub> O, D <sub>2</sub> O, Graphite, None, (Other-specify)
Thermodynamic Cycle	Rankine	Loop-type (# loops), Direct-cycle, Semi-integral, Integral, Pool-type
Secondary Side Fluid	n/a	Forced (# pumps), Natural
Fuel Form	Fuel Assembly/Bundle	Rankine, Brayton, Combined-Cycle (direct/indirect)
Fuel Lattice Shape	Square	H <sub>2</sub> O, He, CO <sub>2</sub> , Na, Pb, PbBi, Molten Salts, (Other-specify)
Rods/Pins per Fuel Assembly/Bundle	92	Fuel Assembly/Bundle, Coated Sphere, Plate, Prismatic, Contained Liquid, Liquid Fuel/Coolant
Fuel Material Type	Oxide	Square, Hexagonal, Triangular, Cylindrical, Spherical, Other, n/a
Design Status	Conceptual	#, n/a
Licensing Status	DCR	Oxide, Nitride, Carbide, Metal, Molten Salt, (Other-specify)

**Table 2: ARIS Parameter Fields (see also Spreadsheet “Data”) for Booklet**

ARIS Parameter	Value	Units or Examples
<i>Plant Infrastructure</i>		
Design Life	60	years
Lifetime Capacity Factor	95	%, defined as Lifetime MWe-yrs delivered / (MWe capacity * Design Life), incl. outages
Major Planned Outages	10-20 days every 12-24 months (refuelling)  25 days every 120 months (major turbine inspection and ISI on reactor vessel and internals)	# days every # months (specify purpose, including refuelling)
Operation / Maintenance Human Resources	~75 total	# Staff in Operation / Maintenance Crew during Normal Operation
Reference Site Design	1	n Units/Modules
Capacity to Electric Grid	270-290	MWe (net to grid)
Non-electric Capacity	Flexible	
In-House Plant Consumption	10-30	MWe
Plant Footprint	8,400	m <sup>2</sup> (rectangular building envelope)
Site Footprint	26,300	m <sup>2</sup> (fenced area)
Emergency Planning Zone	1 (At site boundary)	km (radius)
Releases during Normal Operation	3.3E+1 / 7.32E-4 / 1.08E-1	TBq/yr (Noble Gases / Tritium Gas / Liquids)
Load Following Range and Speed	50 – 100% daily, 0.5% per minute	
Seismic Design (SSE)	0.3	g (Safe-Shutdown Earthquake)

ARIS Parameter	Value	Units or Examples
NSSS Operating Pressure (primary/secondary)	7.2 / n/a	MPa(abs)
Primary Coolant Inventory (incl. pressurizer)	1,820,000	kg
Nominal Coolant Flow Rate (primary/secondary)	1,530 / n/a	kg/s
Core Inlet / Outlet Coolant Temperature	270 / 287	°C / °C
Available Temperature as Process Heat Source	Flexible 100-200	°C
NSSS Largest Component	Reactor Pressure Vessel (RPV)	
- dimensions	26 / 4 / 485,000	m (length) / m (diameter) / kg (transport weight)
Reactor Vessel Material	SA508	
Steam Generator Design	n/a	
Secondary Coolant Inventory	n/a	
Pressurizer Design	n/a	
Pressurizer Volume	n/a	
Containment Type and Total Volume	Dry(single, underground) / 5,600	type / m <sup>3</sup>
Spent Fuel Pool Capacity and Total Volume	8 / ~1,300	years of full-power operation / m <sup>3</sup>
<b><i>Fuel/Core</i></b>		
Single Core Thermal Power	870	MWth
Refuelling Cycle	12-24	months
Fuel Material	UO <sub>2</sub>	
Enrichment (avg./max.)	3.40 / 4.95	%
Average Neutron Energy	3 - 6E+5	eV

ARIS Parameter	Value	Units or Examples
Fuel Cladding Material	Zircaloy-2	
Number of Fuel "Units"	240	Assemblies
Weight of one Fuel Unit	324	kg
Total Fissile Loading (initial)	A reload of 36 bundles would be ~251.5 kg of <sup>235</sup> U	kg fissile material (specify isotopic and chemical composition)
% of fuel outside core during normal operation	n/a	
Fraction of fresh-fuel fissile material used up at discharge	82	%
Core Discharge Burnup	49.5	GWd/MTU
Pin Burnup (max.)	63 (US regulatory limit is 70)	MWd/kgHM
Breeding Ratio	40 to 60%, a large fraction of the Pu bred is burned before the fuel is discharge	Fraction of fissile material bred in-situ over one fuel cycle or at equilibrium core
Reprocessing	None	
Main Reactivity Control	Rods	
Solid Burnable Absorber	B <sub>4</sub> C, Hf, Gd <sub>2</sub> O <sub>3</sub>	
Core Volume (active)	19	m <sup>3</sup> (used to calculate power density)
Fast Neutron Flux at Core Pressure Boundary	TBD	N/m <sup>2</sup> -s
Max. Fast Neutron Flux	2.3E+18	N/m <sup>2</sup> -s
<b><i>Safety Systems</i></b>		
Number of Safety Trains	Active / Passive	% capacity of each train to fulfil safety function

ARIS Parameter	Value	Units or Examples
- reactor shutdown	Redundant and diverse	100% (control rod run in) / 100% (control rod hydraulic scram)
- core injection	2 / None	100% (high pressure injection through CRD system) / (no core injection required for LOCA mitigation)
- decay heat removal	2 / 4	Two 100% trains / Four 100% trains
- containment isolation and cooling	2 / 1	Two 100% trains / 100% (always in service, not separated into trains/divisions)
- emergency AC supply (e.g. diesels)	Not required	2 non-emergency diesels for plant investment protection. The diesels are not required for reactor safety
DC Power Capacity (e.g. batteries)	24-72	hours
Events in which <i>Immediate Operator Action</i> is required	None	
Limiting (shortest) <i>Subsequent Operator Action</i> Time	24	hours (that are assumed when following EOPs)
Severe Accident Core Provisions	IVMR	In-Vessel Melt Retention (IVMR)
Core Damage Frequency (CDF)	$<10^{-7}$	x per reactor-year (based on reference site and location)
Severe Accident Containment Provisions	PARs, filtered venting	
Large Release Frequency (LRF)	$<10^{-8}$	x per reactor-year (based on reference site and location)
<b><i>Overall Build Project Costs Estimate or Range (excluding Licensing, based on the Reference Design Site and Location)</i></b>		
Construction Time (n <sup>th</sup> of a kind)	26	months from first concrete to criticality
Design, Project Mgmt. and Procurement Effort		person-years (PY) [DP&P]
Construction and Commissioning Effort		PY [C&C]



ARIS Parameter	Value	Units or Examples
Material and Equipment Overnight Capital Cost	~\$1B USD first unit ~\$700M nth of a kind	Million US\$(2015) [M&E], if built in USA
Cost Breakdown	%[C&C] / %[M&E]	
- Site Development before first concrete	/	(e.g. 25 / 10 )
- Nuclear Island (NSSS)	/	( 30 / 40 ) ( 20 / 25 ) ( 20 / 10 )
- Conventional Island (Turbine and Cooling)	/	( 5 / 15 ) ( ----- )
- Balance of Plant (BOP)	/	(to add up to 100 / 100)
- Commissioning and First Fuel Loading	/	
Factory / On-Site split in [C&C] effort	Approximately 60/40	Based on cost ratio. This ratio is heavily influenced the existence of a marine off load facility at the site.

# ***1. Plant Layout, Site Environment and Grid Integration***

## **SUMMARY FOR BOOKLET**

### **Site Requirements during Construction**

The reference site for BWRX-300 is entirely confined in a 170 m by 280 m footprint, which includes the plant building, switchyard, cooling tower, site office, parking lot, warehouse, and other supporting facilities. The reactor building extends below grade where the primary containment vessel (PCV) and reactor pressure vessel (RPV) mostly reside. The underground construction of the reactor building minimizes concrete use.

### **Site Considerations during Operation**

A seismic analysis has been performed for a wide variety soil conditions (soft, medium and hard rock). For construction, the site would need deep water pier for barging or heavy haul routes for land transportation. The largest component transported to site is the reactor pressure vessel (RPV) which is 26 m long and 4 m in diameter and weighs 485 metric tons. During normal operation, the maximum acceptable ambient air temperature is and 37.8°C dry bulb (100°F) / 26.1°C (79°F) mean coincident wet bulb. Maximum recommended Inlet Temp for the main Condenser/Heat Exchanger is 37.8°C (100°F).

### **Grid Integration**

BWRX-300 switchyard requirements are minimal and typical and would be met with a breaker and a half, dual high voltage (HV) bus design, and a relay house with controlled

#### **1.1. Site Requirements during Construction**

The reference site for BWRX-300 (Figure 1.1.) is entirely confined in a 170 m by 280 m footprint, which includes the plant building, switchyard, cooling tower, site office, parking lot, warehouse, and other supporting facilities. The plant building has a total envelope of 8,400m<sup>2</sup> and contains 6 distinct spaces, or “buildings” within.

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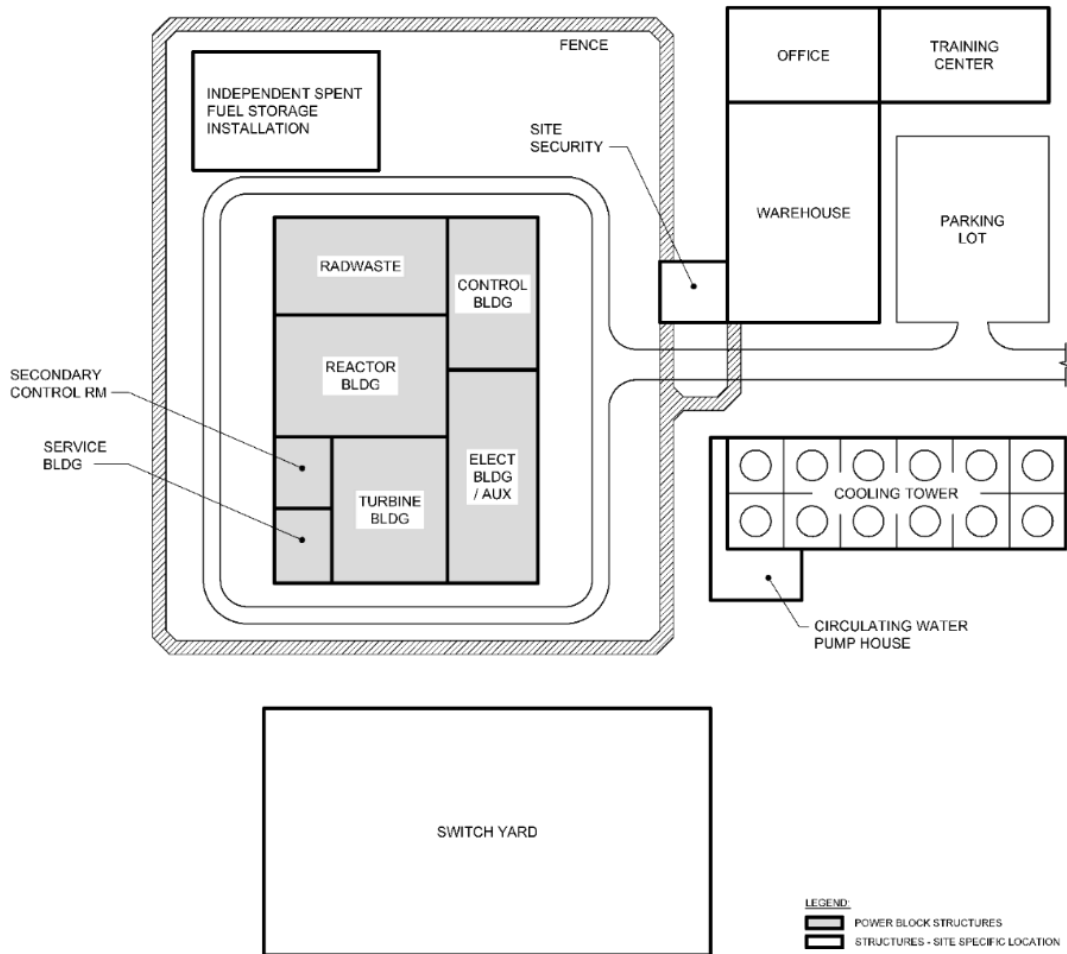


FIG. 1.1. BWRX-300 Plant Layout, Site Plot Plan

The reactor building, a space within the plant building (shown with more detail in Figure 1.2. below along with the turbine building), extends below grade where the primary containment vessel (PCV) and reactor pressure vessel (RPV) mostly reside. A cylindrical pool rests above the PCV and interfaces with the PCV dome. Also, within the reactor building, four separate isolation condenser system (ICS) pools sit next to the pool above the PCV, with one ICS unit in each.

The spent fuel pool is located at grade in the reactor building and has a capacity of 8 years of used fuel and a full core off-load. Since the spent fuel pool is at grade, spent fuel casks can be removed without the use of a heavy crane.

### 1.2. Site Considerations during Operation

The maximum acceptable ambient air temperature is and 37.8°C dry bulb (100°F) / 26.1°C (79°F) mean coincident wet bulb. Maximum recommended Inlet Temp for the main Condenser/Heat Exchanger is 37.8°C (100°F).

The BWRX-300 can be equipped with a wide variety of cooling options, including dry cooling towers and once-through cooling.

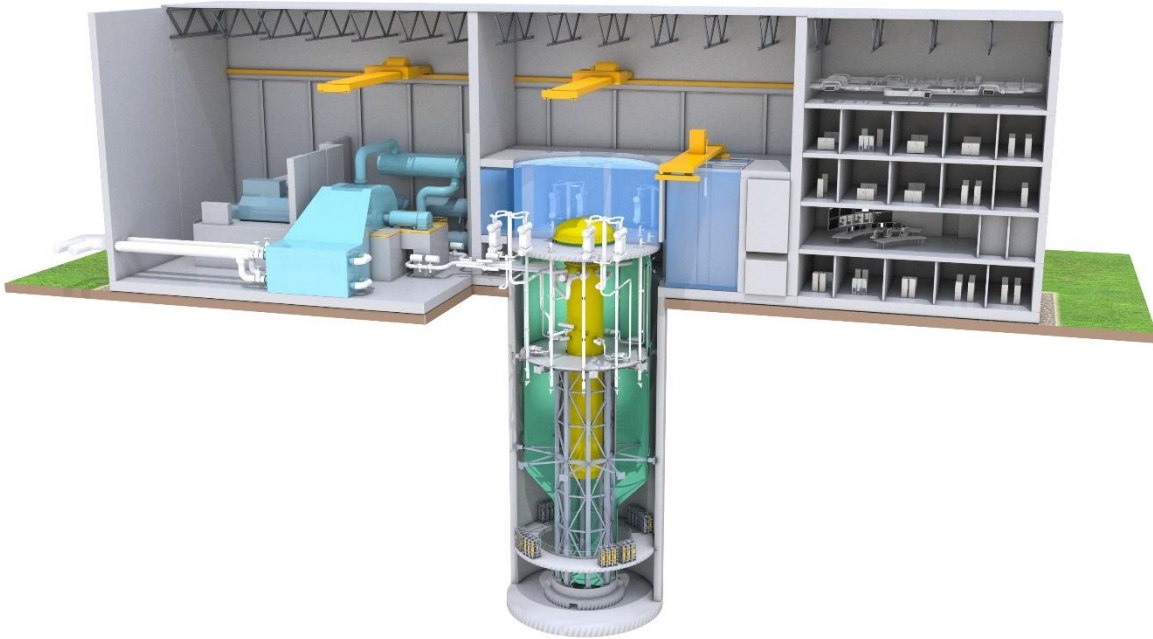


FIG. 1.2. Reactor Building and Turbine Building Cross Sectional View

Normal manpower during operation is 75 people for all shifts.

The baseline option for on-line and outage maintenance is for the plant to have yearly refuelling outages which are 10 to 20 days in duration. This would also include maintenance for key mechanical equipment.

GEH is engaged in the demolition and decommissioning (D&D) business, and GEH intends to incorporate practices into the BWRX-300 design which take D&D costs into account. These practices include using less concrete, as well as using bolts rather than welds wherever possible.

### 1.3. Grid Integration

BWRX-300 switchyard requirements are minimal and typical. BWRX-300 only requires one incoming/outputting transmission line that must be capable of handling the 300 MWe/355 MVA plant output. The BWRX-300 has both 50 and 60 Hz variants. The switchyard should be designed such that no single switchyard breaker or switchyard bus failure results in loss of transmission or loss of plant input power. The switchyard protection schemes should be dual redundant with separate batteries and separate chargers. If required, the plant standby diesel generators can power the switchyard battery chargers. Finally, the various chargers, batteries, and protective relays should be housed in a physically secure and protected enclosure. This equipment would be considered Critical Digital Assets (CDAs) by Federal Energy Regulatory Commission (FERC) in the U.S.

The above requirements would typically be met with a breaker and a half, dual high voltage (HV) bus design, and a relay house with controlled access as shown in Figure 1.4. The instrumentation and controls of the BWRX-300 plant I&C have required interfaces with the switchyard breakers/protective relays; cyber security reasons restrict the interface to hardwired.

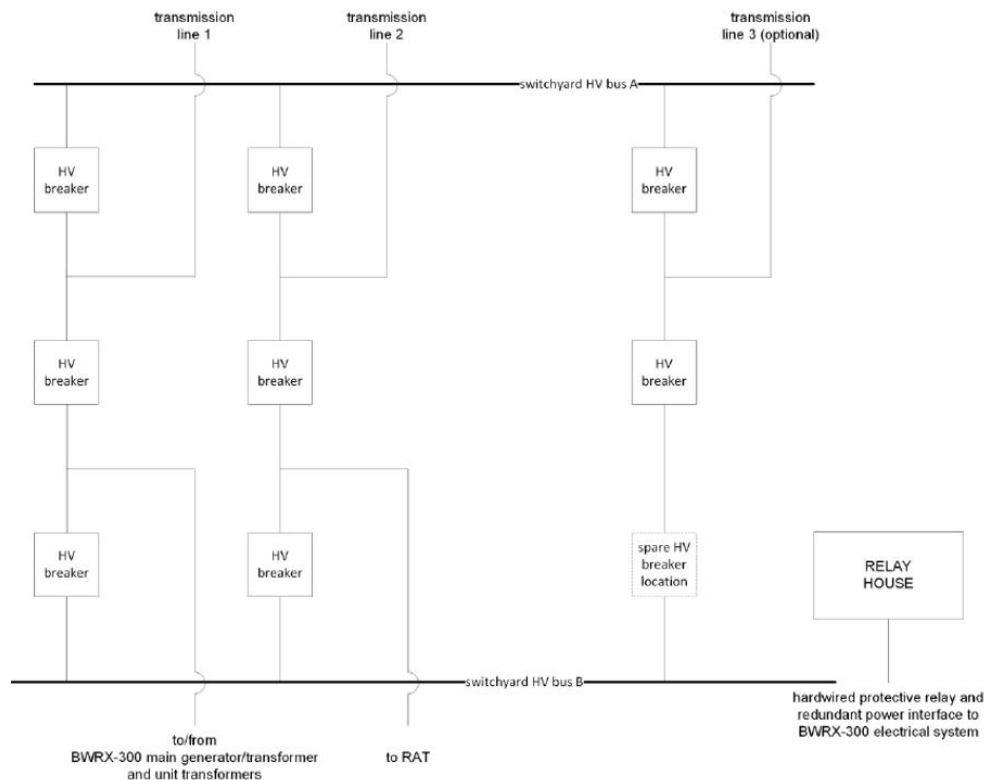


FIG. 1.4. BWRX-300 Switchyard Interface

### **Load following capability, experience, and effect on waste generation**

The BWRX-300 is capable of daily load following to compensate for the effect of variable renewable energy but load following is not the preferred method for frequency control.

### **Ability of plant to operate on house load**

The reference plant is not designed to operate on house load. This can be provided as an option.

### **Ability to load reject without shutdown**

The reference plant is not able to load reject without shutdown. This can be provided as an option.

### **Extend of electrical systems reliance on grid Class IV power**

There is no reliance on grid power for safety functions.

### **Plant needs from the grid (availability, stability, etc.) during normal and off-normal operation**

There is no reliance on grid power for safety functions.

## **Section 1 References**

## ***2. Technical NSSS/Power Conversion System Design***

### **SUMMARY FOR BOOKLET**

#### **Primary Circuit**

BWRX-300 is being designed with the proven supply chain of the ABWR, and predesigned features from the ESBWR, for its primary circuit, or nuclear boiler system. The primary functions of the nuclear boiler system are:

- To deliver steam from the RPV to the turbine main steam system;
- To deliver feedwater from the condensate and feedwater system to the RPV;
- To provide overpressure protection of the reactor coolant pressure boundary (RCPB);
- To provide the instrumentation necessary for monitoring RPV pressure, steam flow, core flow, water level, and metal temperature.

The RPV has an inside diameter of 4 m, a wall thickness of about 136 mm with cladding, and a height of 27 m from the inside of the bottom head (elevation zero) to the inside of the top head. The bottom of the active fuel location is 5.2 m from elevation zero and the active core is 3.8 m high. The relatively tall vessel, due to the chimney, permits natural circulation driving forces to produce abundant core coolant flow.

The major reactor internal components include:

- core (fuel, channels, control rods and instrumentation),
- core support structures (shroud, shroud support, top guide, core plate, control rod guide tube and orificed fuel support),
- chimney
- chimney head and steam separator assembly,
- steam dryer assembly,
- feedwater spargers,
- in-core guide tubes.

#### **Containment/Confinement**

The BWRX-300 Primary Containment Vessel (PCV) is a metal tank that encloses the RPV and is related systems and components. The PCV is dry and is located mostly below grade. The dry PCV is a leak tight gas space surrounding the RPV and the reactor coolant pressure boundary.

#### **Electrical, I&C and Human Interface**

BWRX-300 will use GE equipment for its power conversion systems, namely the STF-D650 steam turbine and the TOPAIR air-cooled generator. The electrical system is a completely integrated power supply and transmission system for the power plant.

The BWRX-300 control and instrument systems provide manual and automatic means to control plant operations and initiate protective actions should plant upset conditions occur. The BWRX-300 utilizes digital controllers, interfacing with plant equipment, sensors and operator controls through a multiplexing system, for signal transmission to achieve these functions.

## 2.1. Primary Circuit

The BWRX-300 is being designed utilizing the proven supply chain of the ABWR, and predesigned features from the ESBWR, for its primary circuit, or nuclear boiler system. Components in this system include the reactor pressure vessel, fine motion control rod drives (FMCRDs), control blades, chimney, separators, and dryer. Although the component designs are the same as those currently deployed in GEH's BWRs, it is important to note the component sizes, specifically the RPV and the chimney height, are scaled appropriately and optimally to the thermal output and natural circulation of the BWRX-300.

The primary functions of the nuclear boiler system are:

- To deliver steam from the RPV to the turbine main steam system;
- To deliver feedwater from the condensate and feedwater system to the RPV;
- To provide natural circulation flow and cooling of the fuel;
- To provide overpressure protection of the reactor coolant pressure boundary (RCPB);
- To provide the instrumentation necessary for monitoring RPV pressure, steam flow, core flow, water level, and metal temperature.

### Reactor pressure vessel

The BWRX-300 reactor pressure vessel (RPV) assembly, shown in Figure 2.1, consists of 1) the pressure vessel, removable head, and its appurtenances, supports and insulation, and 2) the reactor internals enclosed by the vessel (excluding the core, in-core nuclear instrumentation, neutron sources, control rods, and control rod drives). The RPV instrumentation to monitor the conditions within the RPV is designed to cover the full range of reactor power operation. The RPV, together with its internals, provides guidance and support for FMCRDs. Details of the RPV and internals are discussed below.

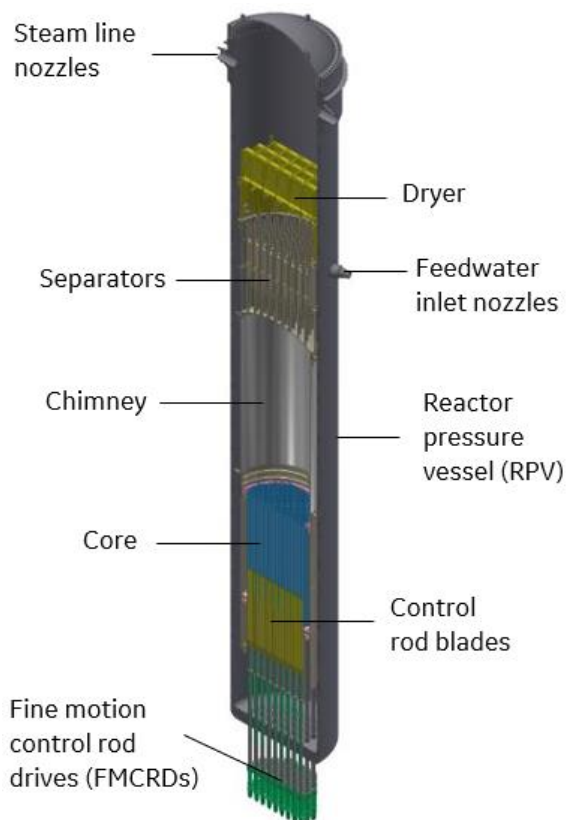


FIG. 2.1. BWRX-300 Reactor Pressure Vessel and Internals

The RPV is a vertical, cylindrical pressure vessel comprising rings and rolled plate welded together, with a removable top head, head flanges, seals and bolting. The vessel also includes penetrations, nozzles, and shroud support.

The reactor vessel has an inside diameter of 4 m, a wall thickness of about 136 mm with cladding, and a height of 27 m from the inside of the bottom head (elevation zero) to the inside of the top head. The bottom of the active fuel location is 5.2 m from elevation zero, and the active core is 3.8 m high. The relatively tall vessel permits natural circulation driving forces to produce abundant core coolant flow.

An increased internal flow path length, relative to forced circulation BWRs, is provided by a "chimney" in the space that extends from the top of the core to the entrance to the steam separator assembly. The chimney and steam separator assembly are supported by a shroud assembly that extends to the top of the core.

### **Reactor internals**

The major reactor internal components include:

- core (fuel, channels, control rods and instrumentation),
- core support and alignment structures (shroud, shroud support, top guide, core plate, control rod guide tube, control rod drive housings, and orificed fuel support),
- chimney
- chimney head and steam separator assembly,
- steam dryer assembly,
- feedwater spargers,
- in-core guide tubes.

Except for the Zircaloy in the reactor core, these reactor internals are stress corrosion resistant stainless steel or other high alloy steels.

The fuel assemblies (including fuel rods and channels), control rods, chimney head and steam separator assembly, steam dryers and in-core instrumentation assemblies are removable when the reactor vessel is opened for refuelling or maintenance. In addition, the internals are designed to be removable.

The RPV shroud support is designed to support the shroud and the components connected to the shroud, including the steam separator, chimney, core plate, and top guide. The fuel bundles are supported by the orificed fuel support, the control rod guide tube and the control rod drive housing. Alignment of the fuel channels is provided by the top guide and the core plate.

### **2.2.Reactor Core and Fuel**

The BWRX-300 core design uses a 240-bundle core configuration. The core design includes [GNF2](#) fuel bundles because of its low hydraulic resistance, which is beneficial for natural circulation. For more information about GNF2 fuel, see Section 5 below. The core lattice configuration (shown in Figure 2.2 below) with equal spacing on the control rod and non-control rod sides (N-lattice) has been chosen for BWRX-300 because it provides more shutdown margin as desired for reload design to accommodate variations in burnup history imposed by load following.



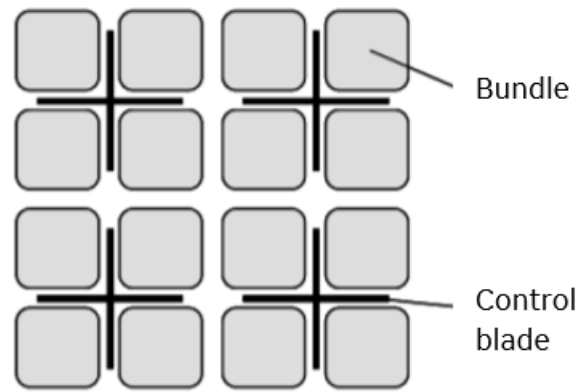


FIG. 2.2. Conventional N-Lattice Design

### 2.3. Fuel Handling

The reactor building is supplied with a refuelling machine for fuel movement and servicing plus an auxiliary platform for servicing operations from the vessel flange level. The refuelling machine is a gantry crane, which spans the reactor vessel and the storage pools on tracks in the refuelling floor. A telescoping mast and grapple suspended from a trolley system is used to lift and orient fuel bundles for placement in the core and/or storage racks.

A position indicating system and travel limit computer are provided to locate the grapple over the vessel core and prevent collision with pool obstacles. The mast grapple has a redundant load path so that no single component failure results in a fuel bundle drop. Interlocks on the machine: (1) prevent hoisting a fuel bundle over the vessel unless an all-control-rod-in permissive is present; (2) limit vertical travel of the fuel grapple to provide shielding over the grappled fuel during transit; and, (3) prevent lifting of fuel without grapple hook engagement and load engagement.

Storage racks are provided for the temporary and long-term storage of new and spent fuel and associated equipment. The new and spent fuel storage racks use the same configuration and prevent inadvertent criticality. Racks provide storage for spent fuel in the spent fuel storage pool in the reactor building. New fuel, 40% of the reactor core, is stored in the new fuel storage vault in the reactor building. The racks are top-loading, with fuel bail extended above the rack. The spent fuel racks have a minimum storage capacity of 298% of the reactor core, which is equivalent to a minimum of 620 fuel storage positions. The new and spent fuel racks maintain a subcriticality of at least 5%  $\Delta k/k$  under dry or flooded conditions. The rack arrangement is designed to prevent accidental insertion of fuel assemblies between adjacent racks and allows flow to prevent the water from exceeding 100°C.

### 2.4. Reactor Protection

The reactor protection system (RPS) is a system of instrument channels, trip logic, trip actuators, manual controls, and scram logic circuitry that initiates the rapid insertion of control rods by hydraulic force to scram the reactor when unsafe conditions are detected.

An alternate method of reactor shutdown from full power to cold subcritical is injection of a neutron absorbing solution to the RPV. Injection initiates as required to mitigate an anticipated transient without scram (ATWS).

The control rod drive (CRD) system comprises three major elements: the fine motion control rod drive (FMCRD) mechanisms; the hydraulic control unit (HCU) assemblies; and the control rod drive hydraulic (CRDH) subsystem.

The FMCRDs (see cross-section in Figure 2.3 below) are designed to provide electric-motor-driven positioning for normal insertion and withdrawal of the control rods and hydraulic-powered rapid control rod insertion (scram) in response to manual or automatic signals from the reactor protection system (RPS). In addition to hydraulic-powered scram, the FMCRDs also provide electric-motor-driven run-in of all control rods as a path to rod insertion that is diverse from the hydraulic powered scram.

The hydraulic power required for scram is provided by high-pressure water stored in the individual HCUs. The HCUs also provide the flow path for purge water to the associated drives during normal operation.

The CRDH subsystem supplies high-pressure demineralized water, which is regulated and distributed to provide charging of the HCU scram accumulators, purge water flow to the FMCRDs, and backup makeup water to the RPV when the feedwater flow is not available. There are 57 FMCRDs mounted in housings welded into the RPV bottom head. Each FMCRD has a movable hollow piston tube that is coupled at its upper end, inside the reactor vessel, to the bottom of a control rod. The piston is designed such that it can be moved up or down, both in fine increments and continuously over its entire range, by a ball nut and ball screw driven by the motor. In response to a scram signal, the piston rapidly inserts the control rod into the core hydraulically using stored energy in the HCU scram accumulator. The FMCRD design includes an electro-mechanical brake on the motor drive shaft and a ball check valve at the point of connection with the scram inlet line. These features prevent control rod ejection in the event of a failure of the scram insert line. There are 29 HCUs, each of which provides sufficient water stored at high pressure in a pre-charged accumulator to scram two FMCRDs at any reactor pressure.

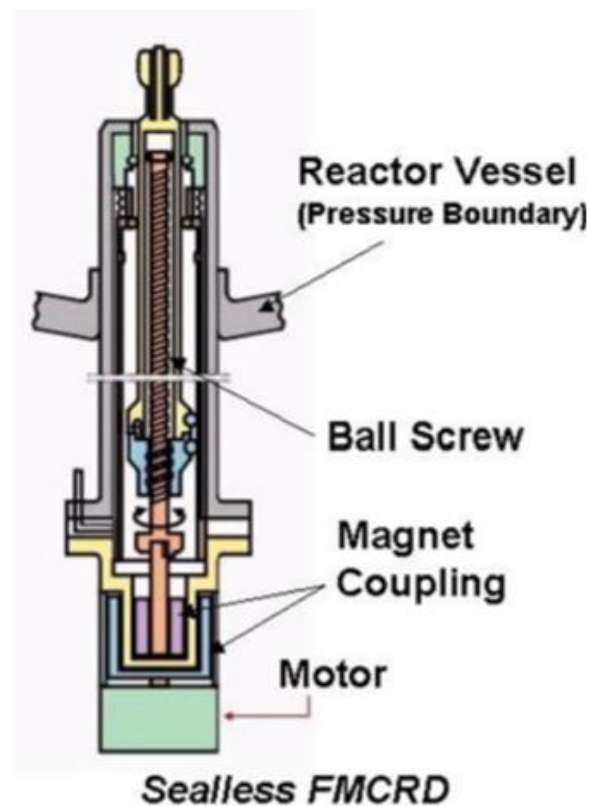


FIG. 2.3. Magnet Coupling Fine Motion Control Rod Drive (FMCRD)

## 2.5. Balance of Plant Process Systems

**Plant service water system (PSWS)** - The PSWS rejects heat from non-safety related components in the reactor and turbine buildings to the environment. The PSWS consists of two independent and 100% redundant open loops continuously re-circulating water through the heat exchangers of the component cooling water system (CCW).

**Component cooling water system (CCW)** - The CCW cools components in the plant and provides a barrier against potential radioactive contamination of the PSWS. The CCW consists of two 100% capacity independent and redundant closed loops.

**Makeup water system (MWS)** - The MWS is designed to supply demineralized water to the various non-safety related systems that need demineralized water and provides water to the Isolation Condenser (IC) pools.

**Condensate storage and transfer system (CSTS)** - The CSTS is a non-safety related system that consist of two 100% pumps and lines taking suction from one storage tank that is the normal source of water for makeup to selected plant systems. The CSTS is also used for storage of excess condensate rejected from the condensate & feedwater systems and the condenser hot well.

**Chilled water system (CWS)** - The CWS consists of two independent, non-safety related, subsystems: the reactor building CWS and the turbine building CWS. The CWS provides chilled water to the cooling coils of air conditioning units and other coolers in the control building and the primary containment. Each subsystem consists of two 100% capacity, redundant, and independent loops with cross-ties between the chilled water piping.

## 2.6. Containment/Confinement

The BWRX-300 Primary Containment Vessel (PCV) is a metal tank that encloses the RPV and is related systems and components. The PCV is dry and is located mostly below grade. The details of the PCV are shown in Figure 2.4. The dry PCV is a leak tight gas space surrounding the RPV and the reactor coolant pressure boundary. It provides confinement of radioactive fission products, steam and water released in the unlikely event of a LOCA. The containment is designed to contain the fission products, steam and water released during a LOCA. Large LOCAs are isolated using RPV Isolation valves.

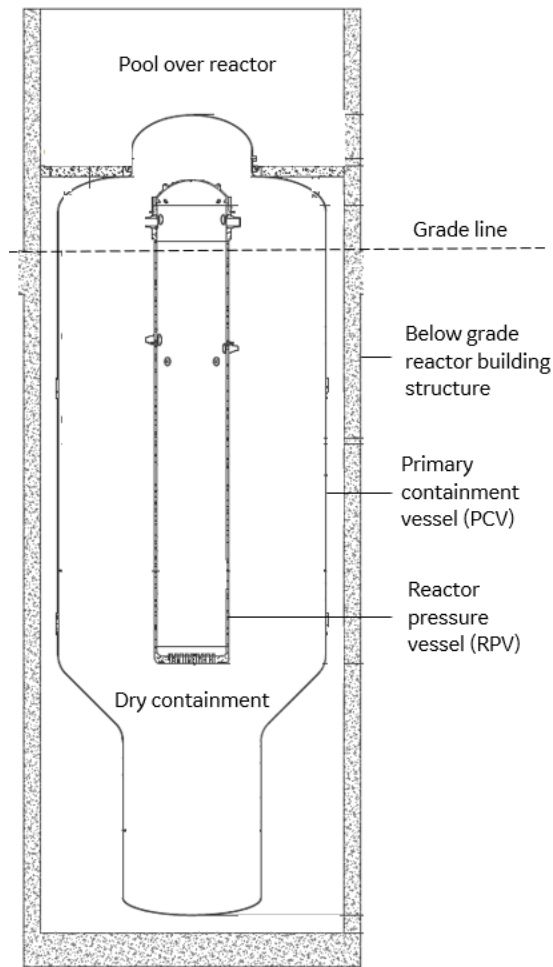


FIG. 2.4. Containment

The high radiation areas (not shown in Figure 2.4 above) are access controlled and locked during normal operation. There is one access point for personnel access to radiation and potential contamination areas. High radiation and contaminated areas will be minimized.

### 2.7. Electrical, I&C and Human Interface

BWRX-300 will use GE's STF-D650, a steam turbine with maximum output of 700 MWe, with reheat, and with double flow in the low-pressure turbine section, see Figure 2.6 for cutaway view and data table. There are 79+ STF-650 units in operation. For more information, see the data sheet for [STF-650 series turbines](#).

	STF-D650
Main Steam	Up to 2,680 psi (185 bar) Up to 1,112°F (600°C)
Reheat Temperature	Up to 1,112°F (600°C)
Frequency	50 Hz and 60 Hz
Output	150 MW–700 MW
Unit Experience	87 Units



FIG. 2.5. GE STF-D650 Steam Turbine

The generator type to be deployed for BWRX-300 is GE's TOPAIR. TOPAIR is an air-cooled generator with operating frequencies of 50 and 60 Hz, depending on geographical location requirements. There are 3,570+ TOPAIR units in operation. For more information, see data sheet for [TOPAIR](#).

	50 Hz	60 Hz
Power Factor	0.80	0.85
Apparent Power	Up to 430 MVA	Up to 360 MVA
Efficiency	Up to 98.9%	Up to 98.8%
Terminal Voltage	Up to 22 kV	Up to 22 kV

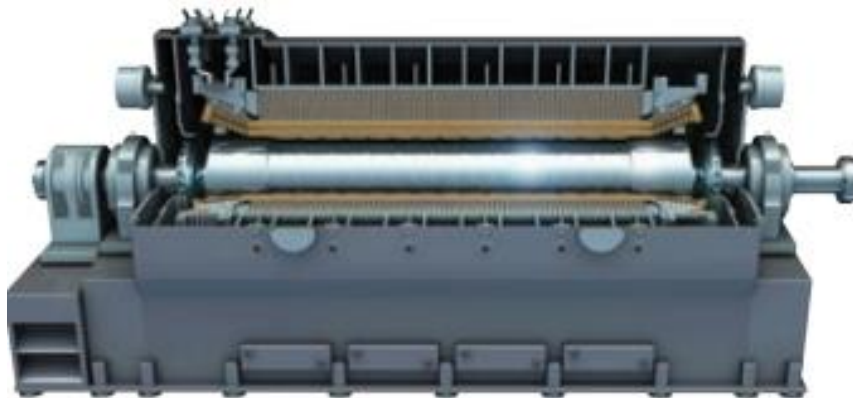


FIG. 2.6. GE TOPAIR Generator

### Electrical systems

The BWRX-300 electrical system is a completely integrated power supply and transmission system for the power plant. It is divided into subsystems based on safety classification.

Each subsystem has appropriate levels of hardware and software quality (corresponding to the systems they power), to provide reliable power to various plant electrical loads, and a transmission path for the main generator to the utility switchyard/grid. The electrical systems

are appropriately protected, monitored, recorded, and controlled by the plant operators. Various segments of the plant electrical system are capable of operating independently.

The function of the highest safety class BWRX-300 electric system is to provide power to the highest safety class Instrumentation & Control (I&C) system and any mechanical components.

### **I&C design architecture and applicable standards**

The BWRX-300 I&C system (also referred to as the Distributive Control and Information System or “DCIS”) is a completely integrated control and monitoring system for the power plant. The I&C comprises three main platforms.

Each platform has appropriate levels of hardware and software quality (corresponding to the systems they control), and provides control, monitoring, alarming and recording functions. The various components of the DCIS are capable of operating independently.

### **Control room layout**

The BWRX-300 control and instrument systems provide manual and automatic means to control plant operations and initiate protective actions should plant upset conditions occur. The BWRX-300 utilizes digital controllers, interfacing with plant equipment, sensors and operator controls through a multiplexing system, for signal transmission to achieve these functions. The key distinguishing simplification features for plant control and monitoring include:

- Enhanced human-system interface design
- Automated plant operations
- Simplified neutron monitoring system
- Reduction in number of nuclear boiler instruments
- Fault-tolerant safety system logic and control
- Standardized digital control and measurement
- Multiplexing of plant control signals

Multiplexed signal transmission, using high-speed fibre optic data links, is combined with digital technology to integrate control and data acquisition for both reactor and turbine plants. Multiplexing significantly reduces the quantities of control cables which need to be installed during construction, thereby reducing the construction cost, and facilitates automation of plant operations.

Performance monitoring and control, and power generator control subsystem functions are provided by the process computer system (PCS) to support efficient plant operation and automation.

The main control room (MCR) panels consist of an integrated set of operator interface panels (e.g., main control console, large display panel). The safety-related panels are seismically qualified and provide grounding, electrical independence and physical separation between safety divisions and non-safety related components and wiring.

The MCR panels and other MCR operator interfaces are designed to provide the operator with information and controls needed to safely operate the plant in all operating modes, including start-up, power operation, refuelling, shutdown, and cold shutdown. Human factors engineering principles have been incorporated into all aspects of the BWRX-300 MCR design.

### **2.8. Unique Technical Design Features (if any)**

The BWRX-300 has several technical design features that are new to boiling water reactor technology.

- 1) RPV Isolation valves. The BWRX-300 RPV is equipped with RPV isolation valves which help mitigate the effects of a Loss of Coolant Accident (LOCA). All fluid pipe systems > 50 mm diameter is equipped with double isolation valves which are integral the RPV.
- 2) The Safety Relief Valves are the most likely cause of a LOCA and have therefore been eliminated from the design. The isolation condenser system (ICS) provides overpressure protection in accordance with ASME BPV code, section III, class 1 equipment. In order to accommodate this change, the design pressure has been raised by 20% from previous boiling water reactors.
- 3) Dry containment – the BWRX-300 has a dry containment. This has been proven to effectively contain the releases of steam, water and fission products after a LOCA.
- 4) Use of commercial off-the-shelf equipment – the BWRX-300 has been designed to use commercial off-the-shelf equipment in the balance of plant (BOP). The turbine and generator have been delivered many times and can be use with small modifications in this plant. This leads to a lower-cost turbine island.

## Section 2 References

1. “GNF2.” <https://nuclear.gepower.com/fuel-a-plant/products/gnf2-advantage>
2. “Status Report 100 – Advanced Boiling Water Reactor II (ABWR II).” Section 2.2. <https://aris.iaea.org/PDF/ABWR-II.pdf>
3. “Status Report 100 – Economic Simplified Boiling Water Reactor.” Sections 2.4, 2.5, 7.3, and 7.4. <https://aris.iaea.org/PDF/ESBWR.pdf>
4. “STF-D650 (Reheat).” <https://www.ge.com/power/steam/steam-turbines/reheat/stfa650>
5. “TOPAIR Air Cooled Generator.” <https://www.ge.com/power/gas/generators/air-cooled-topair>

### ***3. Technology Maturity/Readiness***

#### **SUMMARY FOR BOOKLET**

The BWRX-300 leverages the main features of the power cycle from nuclear power plants that have been or are in service. The BWRX-300 is the 10<sup>th</sup> generation boiling water reactor and draws heavily from previous designs. For example, it leverages natural circulation from the ESBWR and Dodewaard, and utilizes key components from the ABWR. The nuclear core uses the proven GNF2 fuel assemblies that are manufactured and sold to over 80% of the boiling water reactor fleet, and over 18,000 GNF2 fuel assemblies have been delivered worldwide to date (2019).

Here is a brief list of preestablished foundations which support and streamline the BWRX-300 design:

- GEH has more than 20 BWR plants currently in service with hundreds of combined reactor years of operating experience.
- GEH administers and coordinates a Boiling Water Reactor Owners' Group (BWROG) which deals with fleet-wide issues and concerns and operating experience.
- GEH has a US NRC-approved NQA-1 quality program. The quality program is also approved internationally and meets the IAEA requirements for an overall quality program.
- GEH uses the TRACG thermal-hydraulic code to perform design-basis safety analyses. The code was qualified through a series of tests which covered important phenomena with parameter ranges that largely overlap the BWRX-300 parameter range, and has been approved by US-NRC for ESBWR safety analysis.

#### **3.1. Deployed Reactors**

The BWRX-300 leverages the main features of the power cycle from nuclear power plants that have been or are in service. The BWRX-300 is the 10<sup>th</sup> generation boiling water reactor and draws heavily from previous designs. For example, it leverages natural circulation from the ESBWR and Dodewaard, and utilizes key components from the ABWR. The nuclear core is using the proven GNF2 fuel assemblies that are manufactured and sold to over 80% of the boiling water reactor fleet, and over 18,000 GNF2 fuel assemblies have been delivered worldwide to date (2019).

Previous GE BWR designs have been licensed world-wide, including in the US, Japan, UK, Taiwan, Switzerland, Italy and Spain.

GEH provides operational support to more than 20 BWR plants currently in service with hundreds of reactor years of operating experience. In addition to the World Association of Nuclear Operators (WANO) and the US-based Institute of Nuclear Power Operations (INPO), which both coordinate the sharing of nuclear reactor operating experience, GEH also administers and coordinates a Boiling Water Reactor Owners' Group (BWROG) which deals with fleet-wide issues, concerns and operating experience.

GEH has a US NRC-approved NQA-1 quality programme. The GEH quality program is also approved internationally and meets the IAEA requirements for an overall quality program.



GEH is an approved supplier of nuclear quality parts, fuel, and engineering services for many world-wide utilities.

GEH uses the TRACG thermal-hydraulic code to perform design-basis safety analyses of the BWRX-300. The analysis code and methods for each application are described in a series of licensing topical reports covering 1) large-break and small-break LOCA and containment analysis, 2) reactor stability, 3) anticipated operational occurrences, and 4) ATWS. The code was qualified through a series of tests which covered important phenomena with parameter ranges that largely overlap the BWRX-300 parameter range, and has been approved by US-NRC for ESBWR safety analysis. Although different methods are used to evaluate the uncertainty for different analyses, the uncertainty in the calculated safety parameter is evaluated by statistically combining the uncertainties for medium and/or highly ranked PIRT parameters.

### 3.2. Reactors under Licensing Review

The BWRX-300 is undergoing licensing in several different countries. Pre-application activities are underway in the US; Vendor Design Review phase 1 and 2 are underway in Canada; and, in the UK, the BWRX-300 completed a BEIS-funded Mature Technology evaluation. Regulatory activities in other countries will start as required to support commercial activities.

One of the design goals of the BWRX-300 is to be capable of being licensed internationally. The design has been developed with a strict adherence to a philosophy which follows the IAEA Defense-in-Depth guidelines. This creates a transparent, and easy-to-follow foundation tree for plant structure, systems and component classification that can be adapted to different individual country regimes requirements.

An important design goal of the BWRX-300 is to avoid use equipment that is of a new and untested design. All equipment should be already delivered and available commercially to the greatest extent possible.

The BWRX-300 plant goal is to have a 95% capacity factor

The BWRX-300 leverages proven designs and will have very little research and technology development needed. Only certain components will need component-level testing. GEH is also exploring new nuclear manufacturing methods such as those using Powder Metallurgy and advanced design optimization methods. Features such as these will be extensively tested before they are incorporated into the design.

GEH is supported by scientists and engineers at GE Global Research, US universities and US national labs for work in metal manufacturing area and advanced two-phase modelling of the reactor vessel internals, with the goal of optimizing the design. None of these is needed to commercialize the technology but will be leveraged to lower cost in the future.

The current deployment schedule supports a commercial operating date between 2027 and 2028. The exact date will depend on the level of support from potential buyers and the approvals of a nuclear site.

## 4. Safety Concept

### SUMMARY FOR BOOKLET

The basic BWRX-300 safety design philosophy is built on utilization of inherent margins (e.g., larger structure volumes and water inventory) to eliminate system challenges. The first line of defense are the design rules that govern the quality and rigor in the power plant design. The second line of defense is the normal operating system's ability to handle transients and accidents through such design features as adjustable speed, motor driven feedwater pumps and higher capacity CRD pumps with backup power (6.6 kV plant investment protection buses). As a third line of defense, passive safety-related systems are used in the design to provide confidence in the plant's ability to handle transients and accidents.

The primary purpose of the Defense-in-Depth (D-in-D) concept is to establish a singular paradigm through which to view the disparate systems and equipment that perform functions important to the safety case of the BWRX-300 design. The existence of such a paradigm is important to streamline many activities with potential to cause delays and cost overruns. Examples include: 1) achieving agreement among stakeholders on inclusion or exclusion of specific design features; 2) presenting and defending the plant's safety case to a given regulatory authority; and 3) understanding and minimizing the design impacts of licensing a plant within multiple different regulatory frameworks.

There are two passive cooling systems for BWRX-300. The first is the isolation condenser system (ICS) which removes decay heat after any reactor isolation during power operations. Decay heat removal limits further increases in steam pressure and keeps the RPV pressure low. The ICS consists of four independent loops, each containing a heat exchanger, with capacity of 33 MWth, that condenses steam on the tube side and transfers heat by heating/evaporating water in the IC pool, which is vented to the atmosphere.

The other passive cooling system is the passive containment cooling system (PCCS) which removes the decay heat and maintains the containment within its pressure limits for design basis accidents such as a LOCA. It consists of several low-pressure, totally independent heat exchangers suspended in the PCV. The heat in the PCV is transferred to the Reactor pool which is located above the PCV upper head, and which is filled with water during normal operation. The Reactor pool is vented to the atmosphere. PCCS operation requires no sensing, control, logic or power actuated devices for operation. The PCCS condensers are a closed loop integral part of the containment pressure boundary. Since there are no containment isolation valves between the PCCS condensers and the drywell, they are always in "ready standby" mode. The large RPV volume of BWRX-300, as described in Section 2.1 below, provides a substantial reservoir of water above the core, which ensures the core remains covered following transients involving feedwater flow interruptions or loss-of-coolant-accidents (LOCAs).

PSA is used heavily in the development of the design of the BWRX-300. Previous experience and full PSA models for both the ABWR and ESBWR is also leveraged. PSA is used to validate assumptions on frequency of occurrence of postulated initiating events, and to identify complex events and mitigating functions to avoid cliff edge effects as a result from the hazards analyses.

The emergency planning zone is expected to be smaller than traditional large Light Water Reactors. The small plant size and underground construction will lead to a smaller affected zone around a site. The goal is to have the evacuation zone at the site boundary.

#### 4.1. Safety Philosophy and Implementation

The basic BWRX-300 safety design philosophy is built on utilization of inherent margins (e.g., larger structure volumes and water inventory) to eliminate system challenges. The first line of defense are the design rules that govern the quality and rigor in the power plant design. The second line of defense is the normal operating system’s ability to handle transients and accidents through such design features as adjustable speed, motor driven feedwater pumps and higher capacity CRD pumps with backup power (6.6 kV plant investment protection buses). As a third line of defense, passive safety-related systems are used in the design to provide confidence in the plant's ability to handle transients and accidents.

The plant also retains several active systems to handle milder transients. As well, all safety related systems are designed such that no operator actions are needed to maintain safe, stable conditions following a design basis accident. A view of the passive safety system configuration is shown in Figure 4.1 below. Descriptions of some important passive safety related systems are provided in the following section.

#### Defense-in-Depth description

The primary purpose of the Defense-in-Depth (D-in-D) concept is to establish a process for classifying the SSC functions that are important to the safety. This process is important to simplify the many activities with potential to cause delays and cost overruns. Examples include: 1) achieving agreement among stakeholders on inclusion or exclusion of specific design features; 2) presenting and defending the plant’s safety case to a given regulatory authority; and 3) understanding and minimizing the design impacts of licensing a plant within multiple different regulatory frameworks.

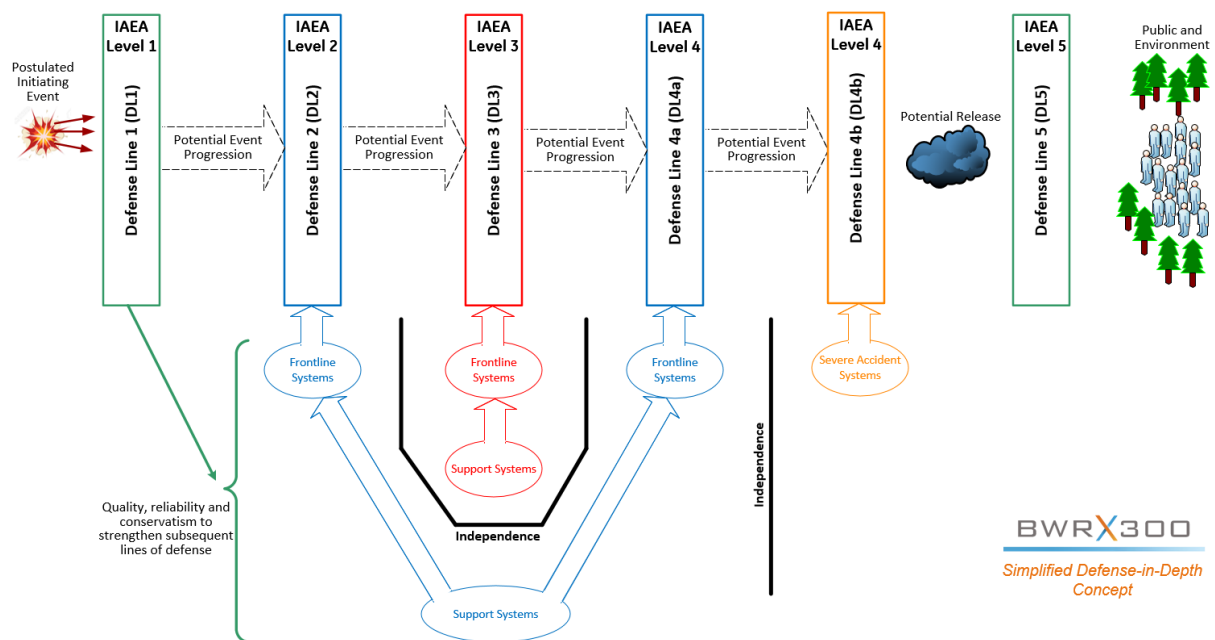


FIG. 4.1. Defense-in-Depth

The BWRX-300 D-in-D concept uses Fundamental Safety Functions (FSF) to define the interface between the defense lines and the physical barriers. In a given plant scenario, if the FSFs are performed successfully, then the corresponding physical barriers will remain effective.

The FSF for the BWRX-300 are:

- Control of reactivity
- Fuel cooling
- Long-term heat removal
- Containment of radioactive materials

In general, 1) the first three FSFs relate to the fuel cladding and RCPB physical barriers, 2) the third FSF also relates to the containment physical barrier, and 3) the fourth FSF corresponds to the RCPB and containment physical barriers.

The BWRX-300 safety assessment framework has been developed to align closely to the D-in-D concept by using a layered deterministic design basis analysis approach.

Note, as shown in Figure 4.1 above, the BWRX-300 line of defense definition diverges from IAEA's at Defense line 4 (DL-4). BWRX-300 separates DL-4 into two independent lines of defense: DL-4a for internal events and DL-4b for external events.

### Isolation condenser system (ICS)

The isolation condenser system (ICS) removes decay heat after any reactor isolation during power operations also provides overpressure protection in accordance with ASME BPV code, Section III, Class 1 equipment. Decay heat removal limits further increases in steam pressure and keeps the RPV pressure low. The ICS consists of four independent loops, each containing a heat exchanger, with capacity of 33 MWth, that condenses steam on the tube side and transfers heat by heating/evaporating water in the IC pool, which is vented to the atmosphere. The arrangement of the IC heat exchanger is shown in Figure 4.2.

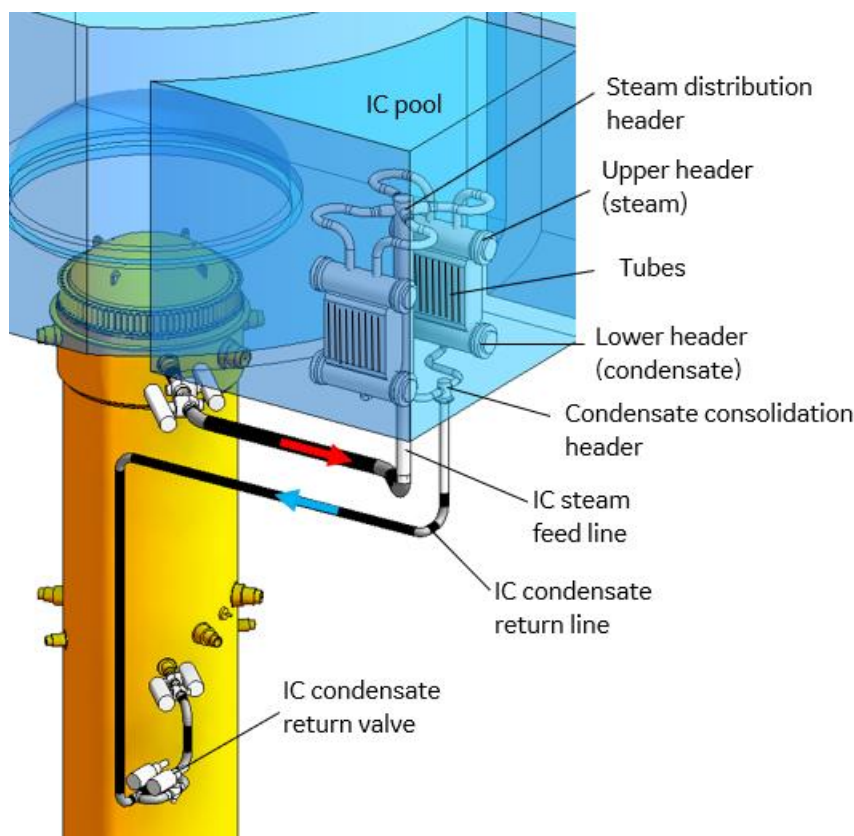


FIG. 4.2. Isolation Condenser System (only 1 train shown)

The ICS is initiated automatically, and the ICS will also be initiated if a loss of DC power occurs. To start an Isolation Condenser (IC) into operation, the IC 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. The ICS can also be initiated manually by the operator from the MCR by opening the IC condensate return valve. The IC pool has an installed capacity that provides 7 days of reactor decay heat removal capability. The heat rejection process can be continued indefinitely by replenishing the IC pool inventory. The ICS passively removes sensible and core decay heat from the reactor when the normal heat removal system is unavailable. Heat transfer from the IC tubes to the surrounding IC pool water is accomplished by natural convection, and no forced circulation equipment is required.

### **Passive containment cooling system (PCCS)**

The PCCS is a passive system that removes the decay heat and maintains the containment within its pressure limits for design basis accidents such as a LOCA. It consists of several low-pressure, totally independent heat exchangers suspended in the PCV. The heat in the PCV is transferred to the Reactor pool which is located above the PCV upper head, and which is filled with water during normal operation. The Reactor pool is vented to the atmosphere. PCCS operation requires no sensing, control, logic or power actuated devices for operation. The PCCS condensers are a closed loop integral part of the containment pressure boundary. Since there are no containment isolation valves between the PCCS condensers and the drywell, they are always in "ready standby" mode.

#### **4.2. Transient/Accident Behaviour**

The large RPV volume of BWRX-300, as described in Section 2.1 above, provides a substantial reservoir of water above the core, which ensures the core remains covered following transients involving feedwater flow interruptions or loss-of-coolant-accidents (LOCAs). This gives an extended period of time during which automatic systems or plant operators can re-establish reactor inventory control using any of several normal, non-safety related systems capable of injecting water into the reactor. Timely initiation of these systems precludes the need for activation of emergency safety equipment.

The large RPV volume also reduces the rate at which reactor pressurization occurs if the reactor is suddenly isolated from its normal heat sink. If isolation should occur, reactor decay heat is rejected to the ICS located within a large pool of water (the IC pool) positioned immediately above (and outside) the containment. Together, the slower pressurization rate and the use of the ICS eliminate the need for relief valves, which is the most likely source of a LOCA for a light water reactor.

The RPV is equipped with RPV Isolation valves for piping systems > 50 mm in diameter. The PCV is a dry containment made from metal.

The severe accident coping mechanisms will include containment filtered vent, AC independent water addition system. Primarily, an in-vessel retention strategies will be used, but the ex-vessel spreadable area and cooling will be provided. Other features will be added to the severe accident system as warranted by the probabilistic safety assessment.

An external hazards evaluation will be performed and the plant will be designed to safely cope with severe external events. The lack of reliance of AC power and the use of ultimate heat sink pools located in the in seismic category 1 reactor building makes the plant design very resilient for impacts from external events.

PSA has been used heavily in the development of the design of the BWRX-300. Previous experience and full PSA models for both the ABWR and ESBWR have also been leveraged. PSA is used to validate assumptions on frequency of occurrence of postulated initiating events, and to identify complex events and mitigating functions to avoid cliff edge effects as a result from the hazards analyses.

The emergency planning zone is expected to be smaller than in traditional large Light Water Reactors. The small plant size and underground construction will lead to a smaller affected zone around a site. The goal is to have the evacuation zone at the site boundary.

**Provisions for safety under external events, e.g. Seismic, fire, flooding, aircraft impact**

Although the ability to predict the magnitude and frequency of beyond-design-basis external events (BDBEE), such as earthquakes and floods, may be improving, and design bases for plants include some margin, some probability will always remain for a beyond-design-basis external event. As a result, though unlikely, external events could exceed the assumptions used in the design and licensing of a plant, as demonstrated by the events at Fukushima.

One of the primary lessons learned from the accident at Fukushima Dai-ichi was the significance of the challenge presented by a loss of safety-related systems following the occurrence of a beyond-design-basis external event.

Additional diverse and flexible strategies that address the potential consequences of these beyond-design-basis external events would enhance safety at each site.

The consequences of postulated beyond-design-basis external events that are most impactful to reactor safety are 1) loss of power and 2) loss of the ultimate heat sink. An approach for adding diverse and flexible mitigation strategies (FLEX) will increase Defense-in-Depth for beyond-design-basis scenarios to address an extended loss of AC power and loss of normal access to the heat sink occurring simultaneously at all units on a site. Figure 4.3 below shows how the diverse coping mechanisms are integrated into the BWRX-300 Defense-in-Depth concept. See reference 2 in the below list of references.



FIG. 4.3. Defense-in-Depth

Defense line 4b consists of the following elements:

- Portable equipment that provides means of obtaining power and water to maintain or restore key safety functions for all reactors at a site.
- Reasonable staging and protection of portable equipment from BDBEEs applicable to a site.
- Procedures and guidance to implement the diverse strategies.
- Programmatic controls that assure the continued viability and reliability of the diverse strategies.

The diverse coping strategies will consist of both 1) an on-site component using equipment stored at the plant site and 2) an off-site component for the provision of additional materials and equipment for longer-term response.

By providing multiple means of power and water supply to support key safety functions, the diverse coping strategies can mitigate the consequences of beyond-design-basis external events.

The objective of the diverse coping strategies is to establish an indefinite coping capability 1) to prevent damage to the fuel in the reactor and spent fuel pools and 2) to maintain the containment function by using installed equipment, on-site portable equipment, and pre-staged off-site resources. This capability will address both 1) an extended loss of AC power (i.e., loss of off-site power, emergency diesel generators and any alternate AC source) and 2) a loss of heat sink which could arise following external events that are within the existing design basis with additional failures and conditions that could arise from a beyond-design-basis external event. Since the beyond-design-basis regime is essentially unlimited, where feasible, plant features and insights from beyond-design-basis evaluations are used to inform coping strategies.

#### **Section 4 References:**

1. “*Status Report 100 – Economic Simplified Boiling Water Reactor.*” Section 3.2.  
<https://aris.iaea.org/PDF/ESBWR.pdf>
2. *NEI 12-06 “Diverse and Flexible Coping Strategies (Flex) Implementation Guide”*, August 2012

## 5. Fuel and Fuel Cycle

### SUMMARY FOR BOOKLET (optional)

The current BWRX-300 fuel design is based on the GNF2 product line because it provides low hydraulic resistance which is desirable for natural circulation. The GNF2 design contains 1) a 10x10 array of 78 full-length fuel rods, 2) 14 part-length rods which span roughly two-thirds of the active core, and 3) two large, central water rods. The core power density of BWRX-300 is lower to maintain greater thermal hydraulic stability.

Because of the large negative moderator density (void) coefficient of reactivity, the BWRX-300 has a number of inherent advantages, including 1) self-flattening of the radial power distribution, 2) spatial xenon stability, and 3) ability to override xenon in order to follow load. The inherent spatial xenon stability of the BWRX-300 permits daily load following over a large core power level range.

The fuel assembly is designed to ensure that fuel damage does not result in the release of radioactive materials in excess of prescribed limits, and that fuel assembly coolability is maintained during postulated accidents. The core nuclear and hydraulic characteristics, plant equipment characteristics, and instrumentation and protection systems are evaluated to assure that those requirements are met.

#### 5.1. Fuel Cycle Options

The reactor lattice configuration and fuel element design for the BWRX-300 are basically the same as employed in operating BWRs around the world. Key features of the BWRX-300 reactor core design are summarized in the below paragraphs.

#### Core Configuration

The reactor core of the BWRX-300 is arranged as an upright cylinder containing fuel assemblies located within the core shroud. The coolant flows upward through the core. The BWRX-300 reactor core comprises fuel assemblies, control rods, and nuclear instrumentation. The fuel assembly and control rod mechanical designs are basically the same as used in all but the earliest GE boiling water reactors; however, evolutionary improvements have been made to these components throughout the history of the GEH BWR. The current generation of these components will be described below for application to the BWRX-300.

#### Fuel Assembly Description

The BWR fuel assembly consists of a fuel bundle and a channel. The fuel bundle contains the fuel rods and the hardware necessary to support and maintain the proper spacing between the fuel rods. The channel is a Zircaloy box which surrounds the fuel bundle to direct the core coolant flow through the bundle and also serves to guide the movable control rods.

The current BWRX-300 fuel design is based on the GNF2 product line. The GNF2 design contains 1) a 10x10 array of 78 full-length fuel rods, 2) 14 part-length rods which span roughly two-thirds of the active core, and 3) two large central water rods. The core power density of BWRX-300 is lower to maintain greater thermal hydraulic stability.

Figure 5.1 below shows the GNF2 design with the major components identified. The cast stainless steel lower tie plate includes a conical section which seats into the fuel support and a grid which maintains the proper fuel rod spacing at the bottom of the bundle. A Defender



debris filter lower tie plate is included to prevent foreign material from damaging the fuel cladding. The cast stainless steel upper tie plate maintains the fuel rod spacing at the top of the bundle and provides the handle that is used to lift the bundle.

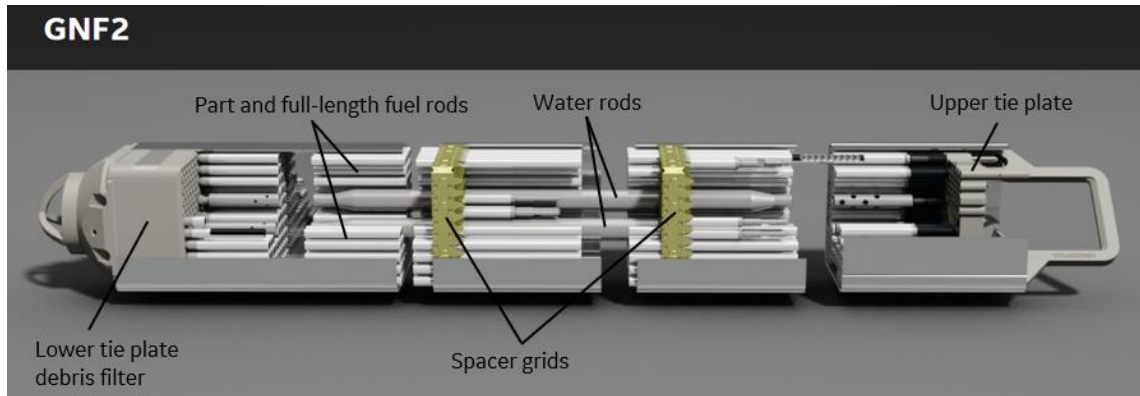


FIG. 5.1. GNF2 Fuel Bundle

The fuel bundle assembly is held together by eight tie rods located around the periphery of the fuel bundle (see Figure 5.2 for cross-sectional view for part, full, tie, and water rod arrangement within a bundle as well as bundle orientation within a four-bundle cell). Each tie rod has a threaded lower end plug which screws into the lower tie plate and a threaded upper end plug which extends through a boss in the upper tie plate and is fastened with a nut. A lock tab washer is included under the tie rod nut to prevent rotation of the tie rod and nut. The part-length rods also have lower end plugs which are threaded into the lower tie plate to prevent movement of the rods during shipping or handling with the bundle oriented horizontally.

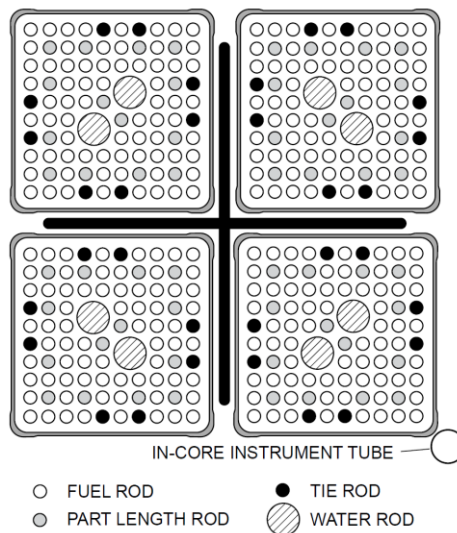


FIG. 5.2. Four Bundle Fuel Module (Cell)

**Section 5 References:**

1. "General Electric Fuel Bundle Designs." NEDO-31152, Revision 9. May 2007.

## 6. Safeguards and Physical Security

### SUMMARY FOR BOOKLET (optional)

#### Safeguards

With respect to safeguards on nuclear material, the BWRX-300 is consistent with operating BWRs. Fuel received on site enters a Special Nuclear Material (SNM) Material Control and Accounting Program, which is the responsibility of the plant operator. The features described here are typical, and the program can be tailored for each individual site. The criteria prescribed in the SNM Material Control and Accounting Program are applicable to SNM and various material mixtures containing SNM. Generally, the SNM involved is plutonium,  $^{233}\text{U}$  or uranium enriched in the isotope  $^{235}\text{U}$ . The  $^{235}\text{U}$  content will vary depending on various reactor parameters. SNM is typically in the form of pellets encapsulated in metal fuel rods. Criteria are established for the SNM control and accounting system, including criteria for the receipt, internal control, physical inventory, and shipment of SNM.

#### Physical Security

With respect to physical security, all vital equipment is located in vital areas to which access is monitored and controlled. Much of the vital area is within radiological control areas which are inaccessible during operation, and typically only accessed during refuelling intervals. Additionally, all vital areas are located within the Protected Area (PA), providing a second physical barrier and means of access control. The “defense in depth” design concepts of redundancy and physical separation of redundant systems further support the physical security of the plant in that multiple vital Structures, Systems or Components (SSC) must be compromised in order to realize effective radiological sabotage. All vital systems and components are housed within robust reinforced concrete structures that are accessed only through a minimal number of normally locked access points that are controlled and monitored by the site security system. Many of the components of vital systems are located below site grade, thereby minimizing exposure to external threats.

#### 6.1. Safeguards

##### Technical features to facilitate implementation of IAEA safeguards

With respect to safeguards on nuclear material, the BWRX-300 is consistent with operating BWRs. Fuel received on site enters a Special Nuclear Material (SNM) Material Control and Accounting Program, which is the responsibility of the plant operator/management. As such the features are described here are typical, and the program can be tailored for individual site.

The criteria prescribed in the SNM Material Control and Accounting Program are applicable to SNM and various material mixtures containing SNM. Generally, the SNM involved is plutonium,  $^{233}\text{U}$  or uranium enriched in the isotope  $^{235}\text{U}$ . The  $^{235}\text{U}$  content will vary depending on various reactor parameters. SNM is typically in the form of ceramic pellets encapsulated in metal fuel rods. Criteria are established for the SNM control and accounting system, including criteria for the receipt, internal control, physical inventory, and shipment of SNM. The remainder of this section provides key portions of a typical program.

Written procedures are prepared and maintained covering the SNM control and accounting system. These procedures shall address, as a minimum, the following topics:

1. Organization and personnel responsibilities and authorities;
2. Designation and description of Item Control Areas (ICAs);
3. Material control records and reporting;

4. Notification for events concerning SNM;
5. Receiving and shipping SNM;
6. Internal transfer of SNM;
7. Physical inventory of SNM;
8. SNM element and isotopic calculation method; and
9. Characterization and identification of items as SNM or non-SNM to preclude loss of control of SNM items.

Shipment procedures are established, to provide for:

1. Verification and recording of the serial number or unique identifier of each item containing SNM;
2. Recording of the quantities of SNM contained in each item;
3. Reporting the quantity of SNM shipped, if the quantity is reportable;
4. Verification of compliance with regulations, including licensing, transportation, and security requirements for shipment; and
5. Reporting the completion of each shipment to the accounting group

Care is taken to assure that SNM contained in fuel is not shipped inadvertently with shipments of nonfuel SNM waste.

The shipment of fuel assemblies, fuel components, or non-fuel SNM is documented in the material control records and the book inventory updated for the applicable ICA. Nuclear Material Transaction Reports are completed.

Records are created and retained in accordance with SNM accountability requirements.

When a fuel assembly has been received by the utility, the SNM custodian takes over SNM accounting responsibilities of the fuel.

## 6.2. Security

The Security Plan of BWRX-300 will be withheld from public disclosure. This section provides general and partial information on physical security features which generally apply to BWRX-300 and to other new nuclear plants.

All vital equipment is located in vital areas to which access is monitored and controlled. Much of the vital area is within radiological control areas which are inaccessible during operation, and typically only accessed during refuelling intervals. Additionally, all vital areas are located within the Protected Area (PA), providing a second physical barrier and means of access control. The Defense-in-Depth concepts of redundancy and physical separation of redundant systems further support the physical security of the plant in that multiple vital Structures, Systems or Components (SSC) must be compromised in order to realize effective radiological sabotage. All vital systems and components are housed within robust reinforced concrete structures that are accessed only through a minimal number of normally locked access points that are controlled and monitored by the site security system. Many of the components of vital systems are located below site grade, thereby minimizing exposure to external threats.

Security Site physical protection is provided through a combination of a Security Organization including armed personnel, physical barriers, controlled access to the PA, controlled access to vital areas located within the PA, and administrative policies and procedures for screening and monitoring personnel and material allowed access to the site.

## 6.3. Unique Safeguards and/or Security Features (if any)

None

**Section 6 References:**

*“ESBWR Design Control Document. Tier 2. Chapter 13: Conduct of Operations.”*  
<https://www.nrc.gov/docs/ML1410/ML14100A541.pdf>

## ***7. Project Delivery and Economics***

### **SUMMARY FOR BOOKLET (optional)**

#### **Project Preparation and Negotiation**

GE-Hitachi Nuclear Energy (GEH) will leverage its and its GE parent's global knowledge and supply chain to deploy the BWRX-300 to those countries having Nuclear Cooperation Agreements in place with the US Government. Project-specific contract terms are anticipated to be negotiated on an individual basis. The base design, and therefore the base cost estimate, of the BWRX-300 is intended to cover a variety of sites; however, individual site parameters and characteristics such as seismicity, restrictions on water usage, local infrastructure and availability of skilled labor will impact the cost for a specific project.

Training programs, with supporting simulators, that are compliant with international and country specific standards will be deployed in parallel with physical plant construction. Additional training can be made available for customers new to nuclear.

#### **Construction and Commissioning**

The BWRX-300's design philosophy of "simplicity" and "designed for constructability" enables it to be constructed and commissioned within an anticipated duration of 30 months from the time of first safety related concrete. For deployment in countries that are new to nuclear, nuclear experienced EPCs are available to team with GEH and local EPCs to implement the project. Local supply of commodities is encouraged to minimize the cost of deployment. However, overall supply chain localization depends on the capability, readiness and aspirations of the receiving country, along with considering project cost vs. localization.

The FOAK BWRX-300 is anticipated to have an overnight EPC cost of \$1B USD or less. As lessons learned from the initial units are gained, NOAK cost are expected drop to approximately \$2,250 USD/kW or less.

#### **Operation and Maintenance**

Achieving a competitive levelized cost of electricity (LCOE) is a primary objective of the BWRX-300 design. It is intended to minimize the required on-site staff and operations and maintenance costs, along with EPC costs. These design, staffing and maintenance objectives are anticipated to produce a LCOE of \$35 to \$50 USD/MWh, depending on a variety of factors including financing costs. GEH can work with potential customers to assist in modelling a project specific LCOE.

## 7. Project Delivery and Economics

### 7.1. Project Preparation and Negotiation

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### 7.2. Construction and Commissioning

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The FOAK BWRX-300 is anticipated to have an overnight EPC cost of \$1B USD or less. As lessons learned from the initial units are gained, NOAK cost are expected drop to approximately \$2,250 USD/kW or less. Export credit agency (ECA) financing should be considered for those projects eligible for this financing, as ECA financing is likely to significantly lower a project's financing costs.

### 7.3. Operation and Maintenance

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