Chapter 3

BOILING WATER REACTORS

3/13/2014

3.1 INTRODUCTION

Boiling Water Reactors (BWRs) operate at a reactor vessel pressure of 1,040 psia, a value considerably lower at almost one half that the operating pressure of PWRs at 2,250 psia. Their nuclear steam supply system is different than PWRs in that the steam is produced within the core and is directly fed to the turbine-generator plant. This system uses a direct cycle as opposed to the indirect cycles used in other nuclear power plants.

BWRs are the most commonly deployed design after the PWR design. The Clinton BWR plant using an artificial cooling lake in Central Illinois, USA is shown in Fig. 1. Other plants use cooling towers.

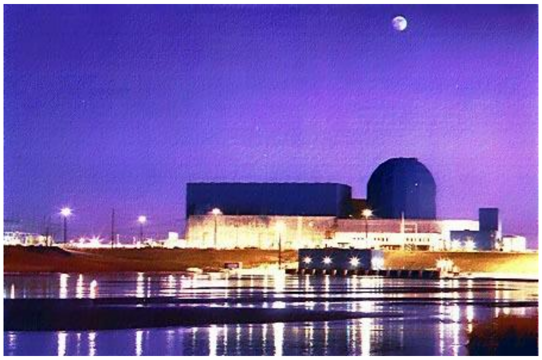


Figure 1. The Clinton Boiling Water Reactor with its artificial cooling lake in Central Illinois, USA.

Figure 2. Susquehanna BWR using cooling towers.

Figure 3. Interior and base view of cooling tower shell.

3.2 BOILING WATER REACTOR POWER CYCLE

The BWR reactor has evolved over time through multiple designs with different features making it more reliable and safer as shown in Table 1.

Table 1. Evolution of BWR reactors designs.

Model	Year of Introduction	Design Features	Typical plants
BWR/1	1955	Natural circulation(Humboldt Bay, Dodewaard) First internal steam separation Isolation condenser Pressure suppression containment	Dresden 1 Big Rock Point Humboldt Bay Dodewaard
BWR/2	1963	Large direct cycle	Oyster Creek
BWR/3/4	1965/1966	First jet pump application Improved Emergency Core Cooling System, ECCS; spray and flood Reactor Core Isolation Cooling, RCIC system	Dresden 2 Browns Ferry
BWR/5	1969	Improved ECCS systems Valve recirculation flow control	LaSalle Nine Mile Point 2
BWR/6	1972	Improved jet pumps and steam separators Reduced fuel duty; 13.4 kW / ft, 44 kW / m Improved ECCS performance Gravity containment flooder Solid-state nuclear system protection system (Option, Clinton only) Compact control room option.	Clinton Grand Gulf Perry
ABWR	1996	Fine motion control rod drives Reactor internal pumps	Kashiwazaki-Kariwa 6,7 Hamaoka 5
ESBWR	-	Gravity flooder Isolation condenser Passive containment cooling Natural circulation	-

Boiling Water reactors with a direct cycle offer the capital cost advantage of eliminating the need for steam generators and a pressurizer. Another feature is the heat transfer in the core is mostly by latent heat as opposed to sensible heat in other power plants. This results in smaller flow rates and pumping energy needs.

These advantages are countered by the presence of the short-lived isotope N^{16} in the turbine plant area. $Nitrogen^{16}$ has a half-life of 7.1 seconds with beta emissions at energies of 4.3 and 10.42 MeV, and Gamma emissions at 6.129 and 7.115 MeV. This prevents access to parts of the turbine area for maintenance until the system is shutdown to allow for N^{16} decay. Additional shielding for the piping, turbine and feed water heaters is needed. Design features which provide a long transit time for the steam to move in the main steam pipe to the top of the reactor building then to the bottom, before getting into the turbine building alleviate the problem. Skyshine of the gamma radiation on the outside of the turbine hall could still be present during operation.

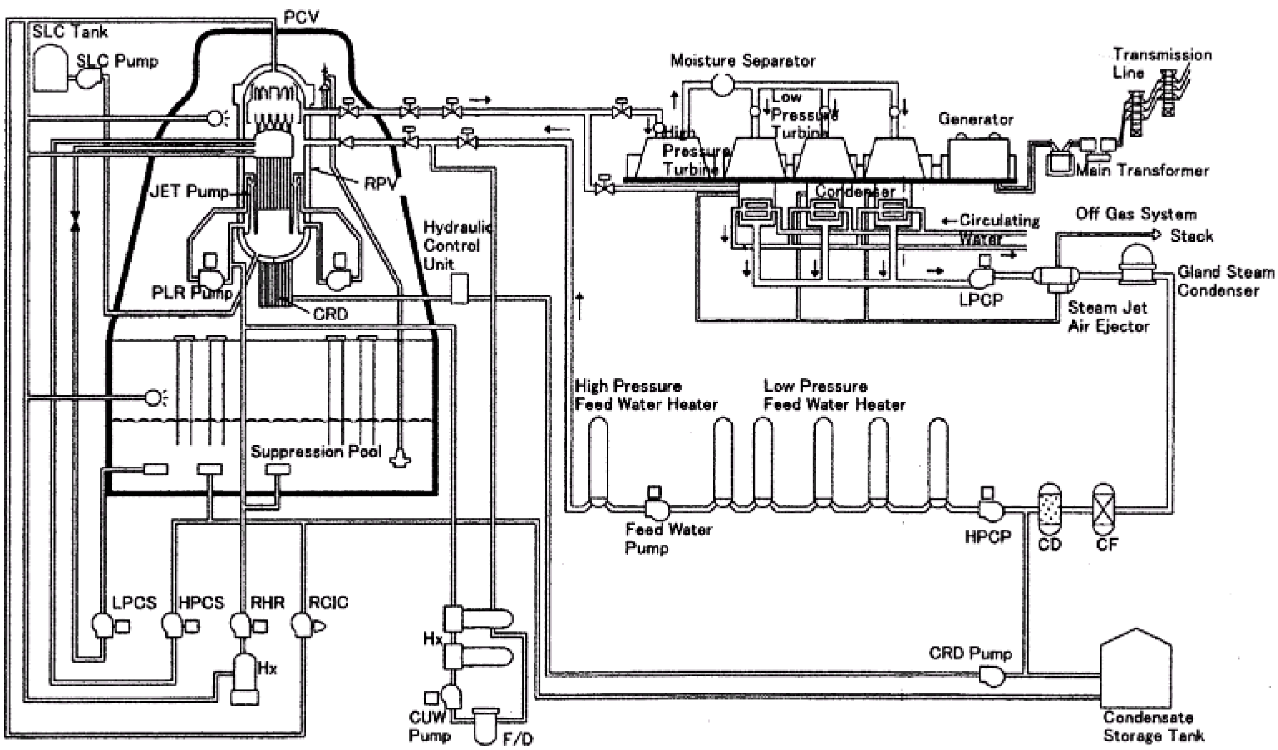


Figure 4. Power cycle of the BWR-5 plant.

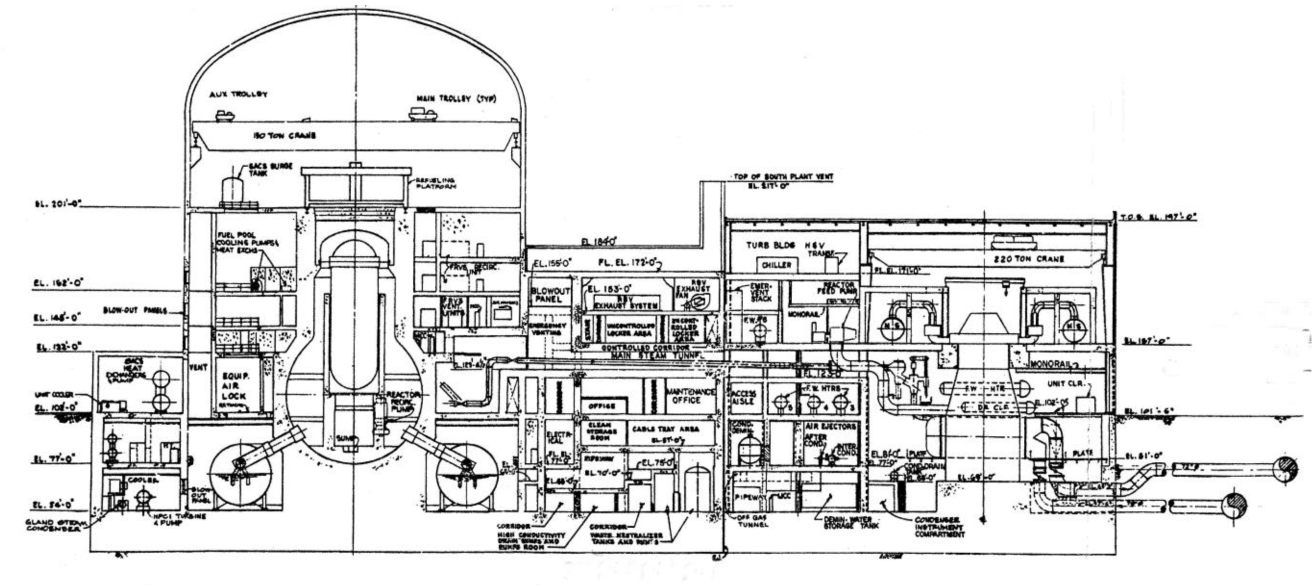


Figure 5. Cross section through BWR and turbine plant showing the below grade components.

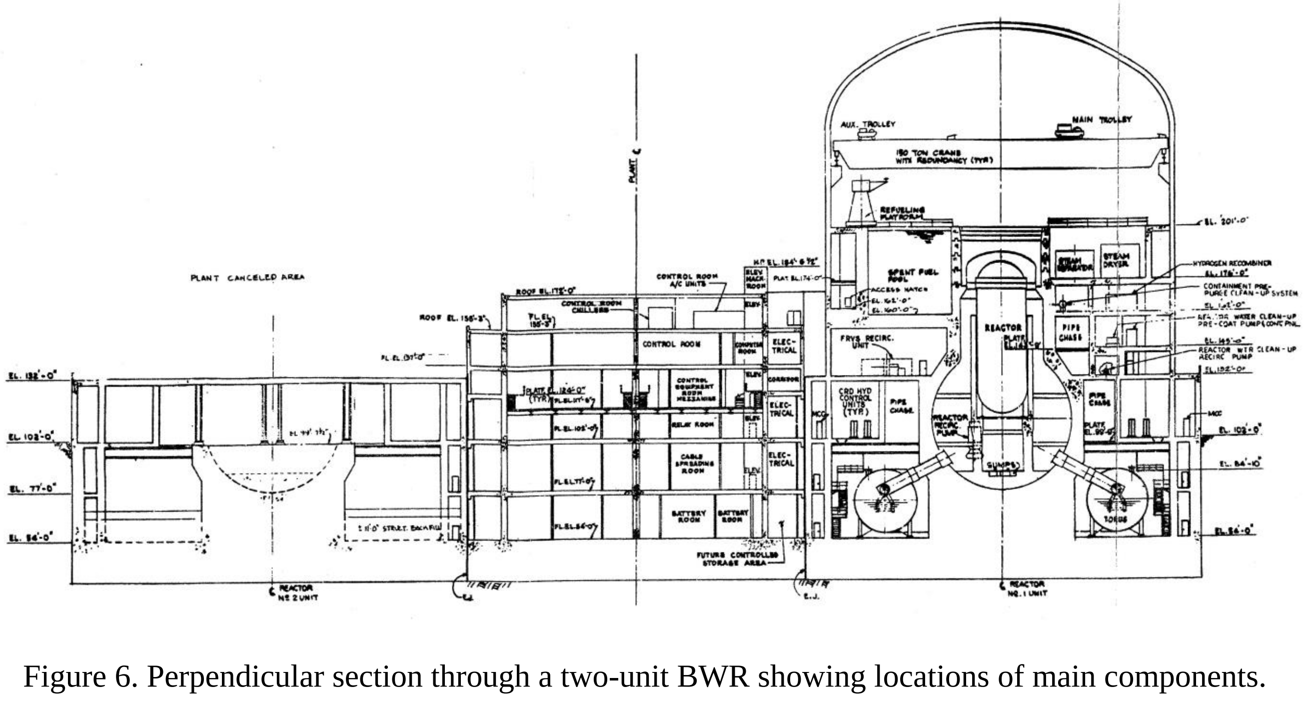


Figure 6. Perpendicular section through a two-unit BWR showing locations of main components.

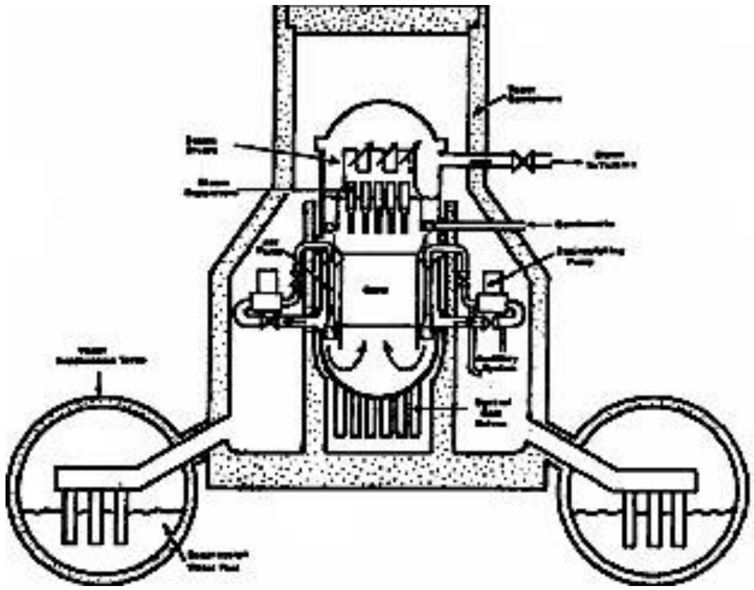


Figure 7. Light Bulb BWR Mark I concrete containment structure showing the pressure suppression pool.

The coolant passing the reactor core requires strict water chemistry management to avoid the activation of the carried corrosion products. The activated corrosion products would otherwise accumulate in piping hot spot, increasing the radiation dose to the operating personnel.

The coolant passing through the core finds 10 percent of it converted into steam. The wet steam goes through a set of steam dryers and separators above the core, which return the separated water to the core. Circulation pumps return the coolant back to the core. The steam is passed from the top of the core to the steam turbine. Because of the presence of the steam drying equipment at the top of the reactor vessel, the control rods in the BWR are inserted from the bottom of the core, in contrast to the PWR system where they are inserted from the top. They contain cooled tubes with boron carbide (B_4C) as a neutron absorber. The control rods are hydraulically driven by the reactor coolant pressure.

Another reason for the control rods to be inserted from the bottom is that they possess there a higher worth since the void fraction is lower near the bottom of the core resulting in higher neutron moderation, higher neutron flux, and higher power level. One would like to control the power level where it peaks closer to the bottom of the core.

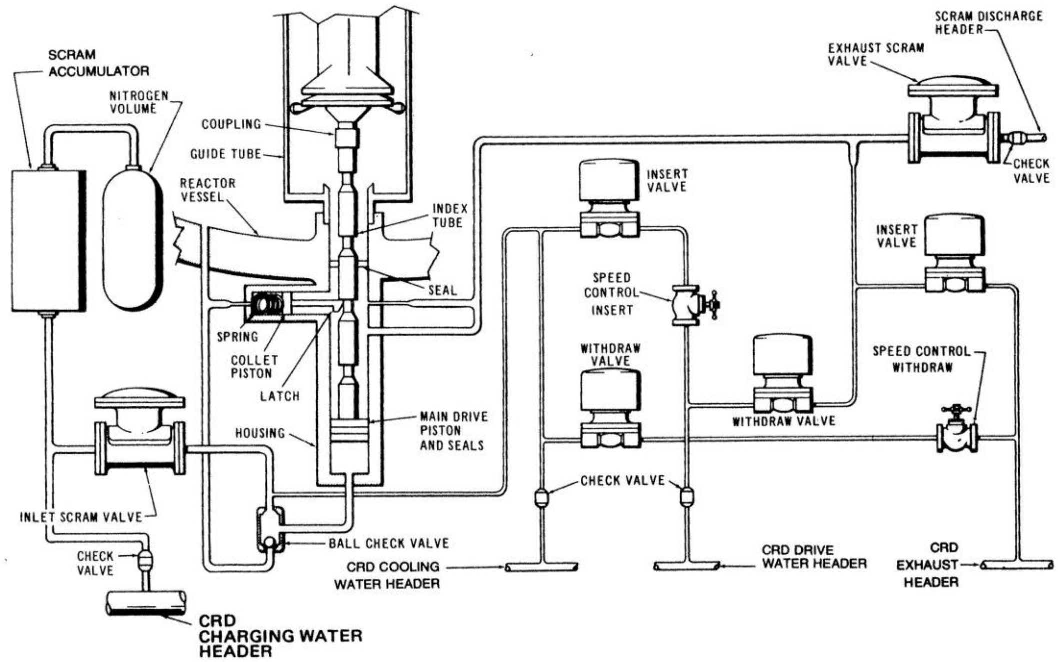


Figure 8. Hydraulically-driven BWR control rods drive mechanism.

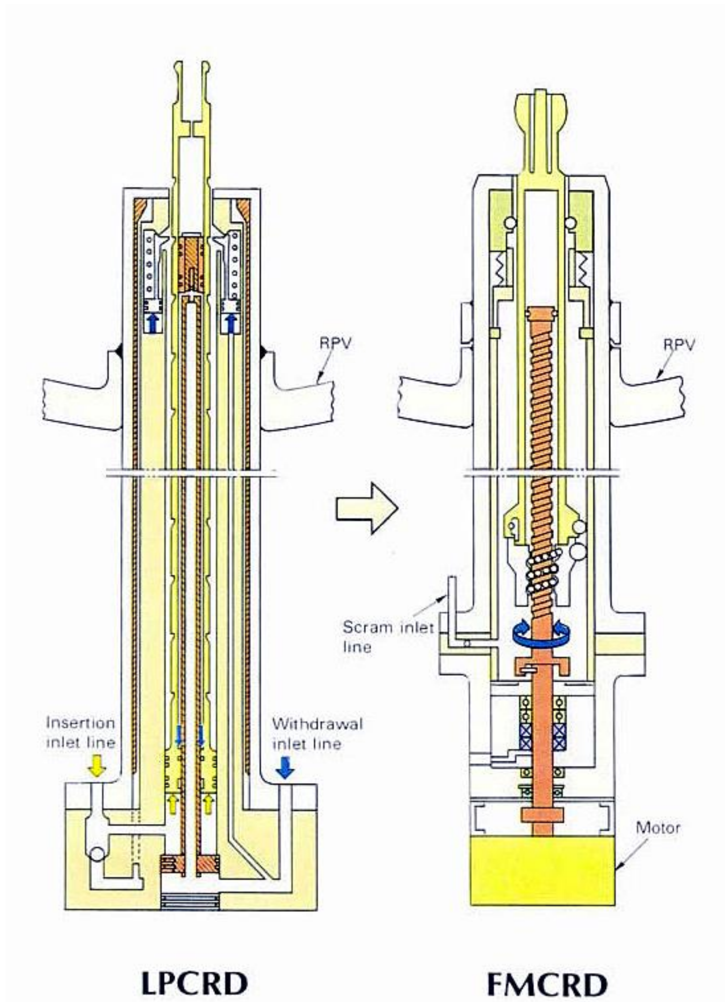


Figure 9. Evolution from hydraulically-driven to linear induction motor-driven control rod drives for the BWR. RPV: Reactor Pressure Vessel. Source: Tepco.

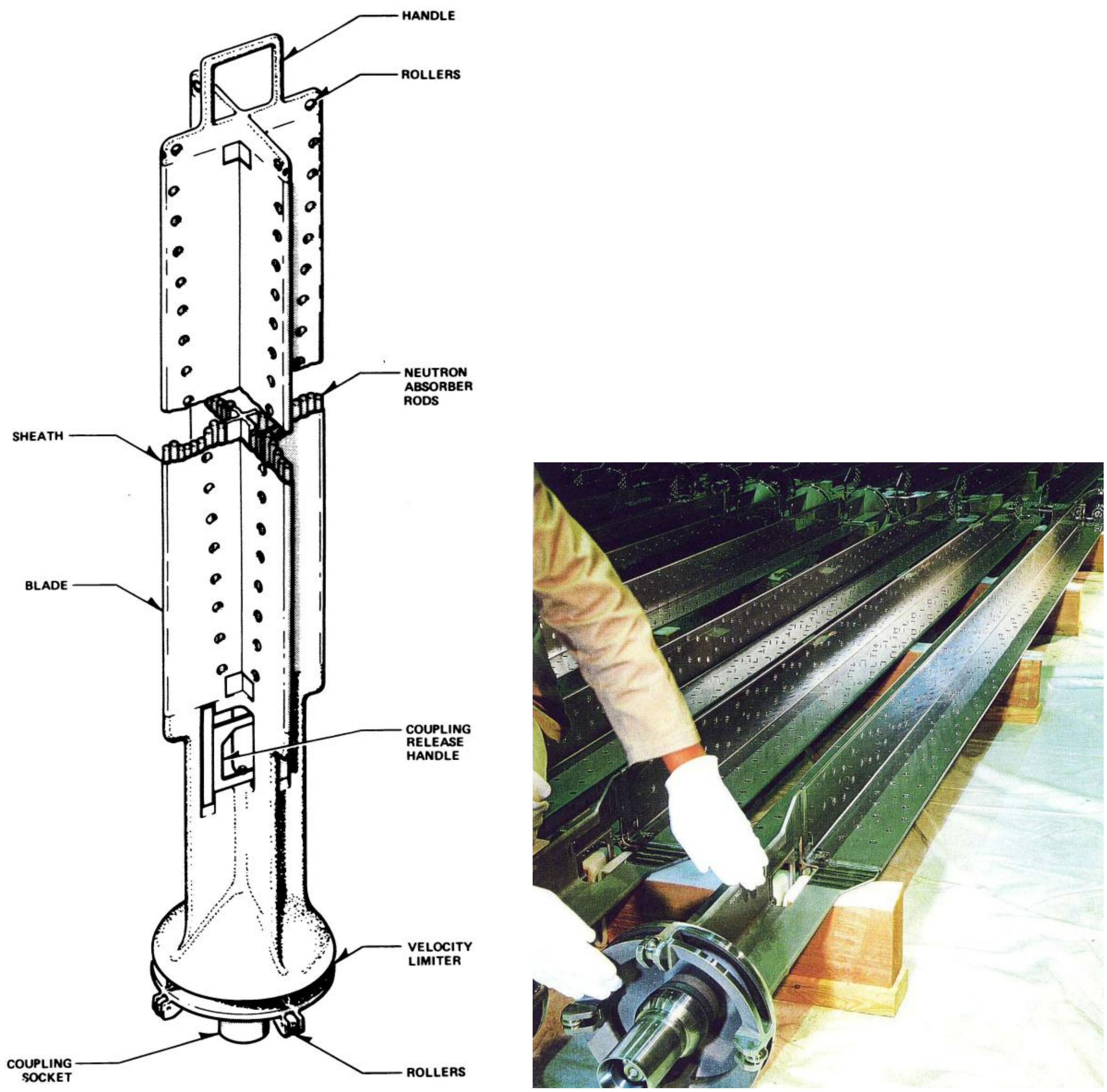


Figure 10. BWR control rod drive. Source: Tepco.

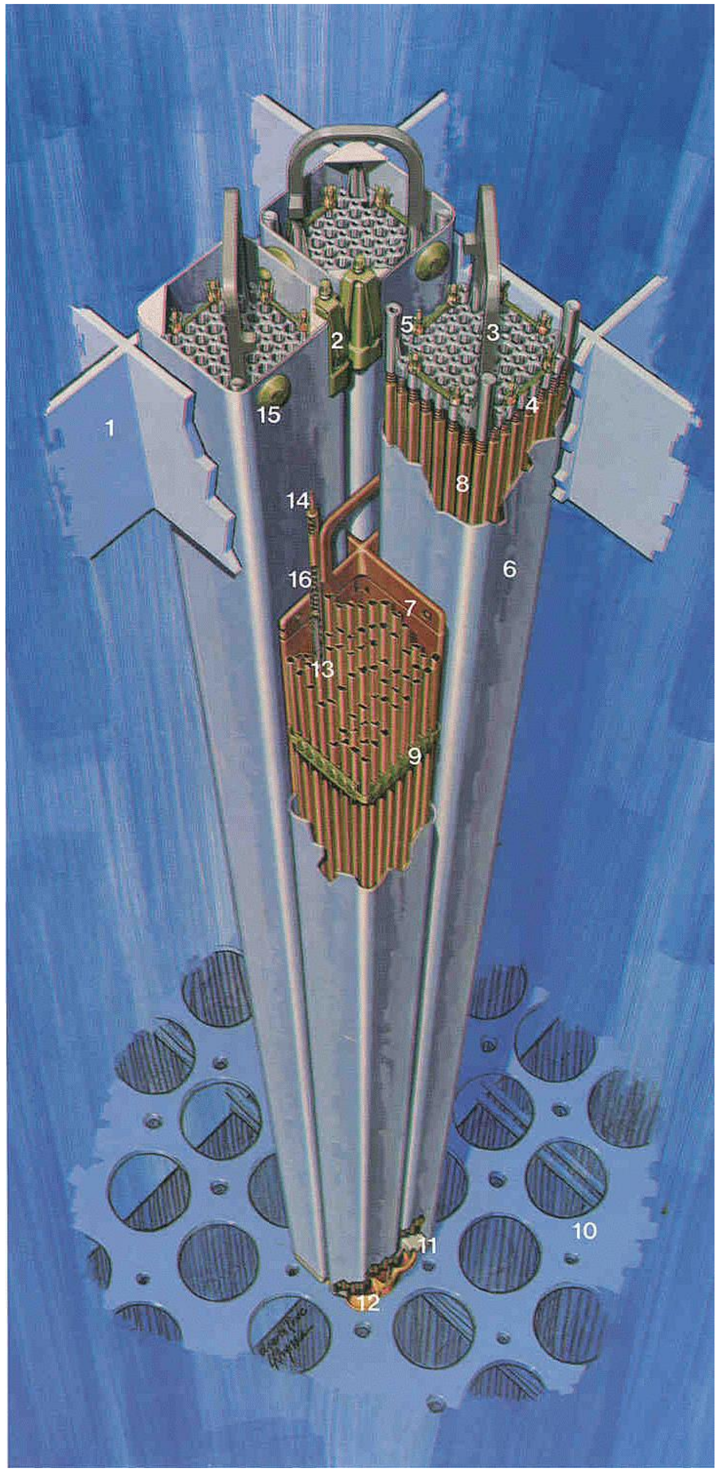


Figure 11. BWR/6 Control rod and fuel assembly configuration. Source: GE.
1: Top fuel guide, 2: Channel fastener, 3: Upper tie plate, 4: Expansion spring, 5: Blocking tab, 6: Channel, 7: Control rod, 8: Fuel rod, 9: Spacer, 10: Core plate assembly, 11: Lower tie plate, 12: Fuel support piece, 13: Fuel pellets, 14: End plug, 15: Channel spacer, 16: Plenum spring.

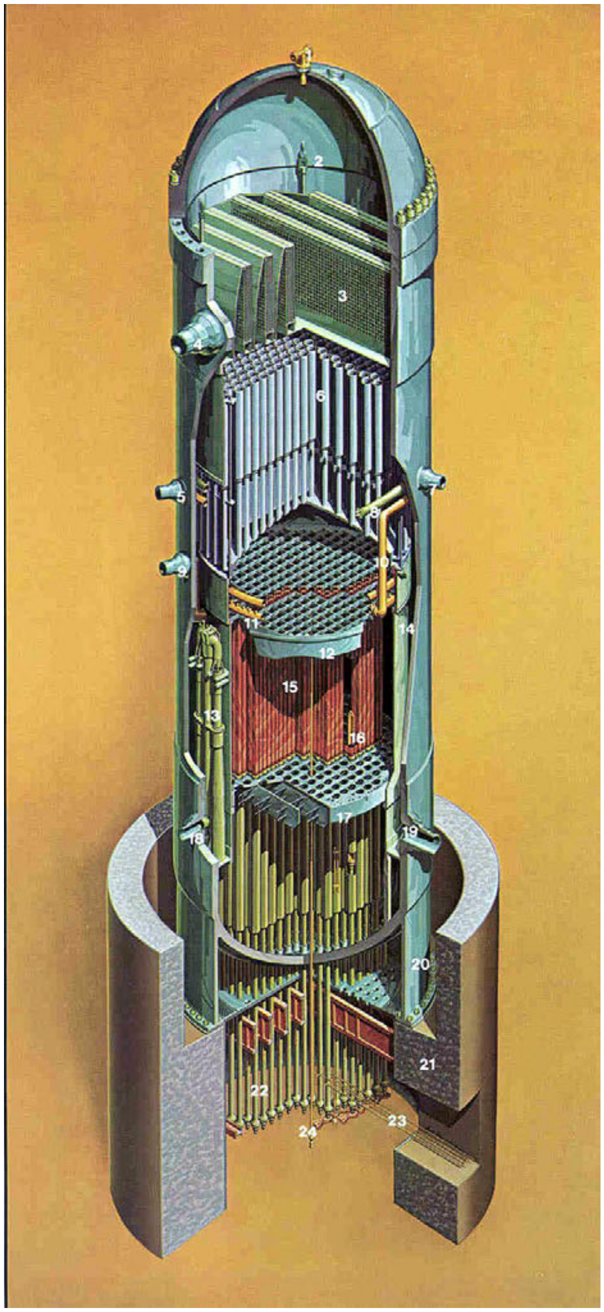


Figure 12. BWR/6 cutout showing core, steam dryers and control rod assembly. Source: GE. 1: Vent and head spray, 2: Steam dryer lifting lug, 3: Steam dryer assembly, 4: Steam outlet, 5: Core spray inlet, 6: Steam separator assembly, 7: Feedwater inlet, 8: Feedwater sparger, 9: Low pressure coolant injection inlet, 10: Core spray line, 11: Core spray sparger, 12: Top guide, 13: Jet pump assembly, 14: Core shroud, 15: Fuel assemblies, 16: Control rod blade, 17: Core plate, 18: Jet pump / recirculation water inlet, 19: Recirculation water outlet, 20: Vessel support skirt, 21: Shield wall, 22: Control rod drives, 23: Control rod drive hydraulic lines, 24: In-core flux monitor.

3.3 CONTAINMENT STRUCTURE

The BWR vessel is surrounded in a containment structure equipped with a pressure suppression pool shown for a light-bulb containment design, and for a steel shell and concrete containment design.

The positioning of the pressure suppression pool below the reactor core does not allow for natural circulation convective cooling in the case of a loss of coolant accident (LOCA). More advanced inherently safe designs position the pressure suppression pool above the core. In case of an accident, the pressure in the core and the pressure suppression pool are equalized by the operator allowing natural circulation from the core to the pressure suppression pool in this case.

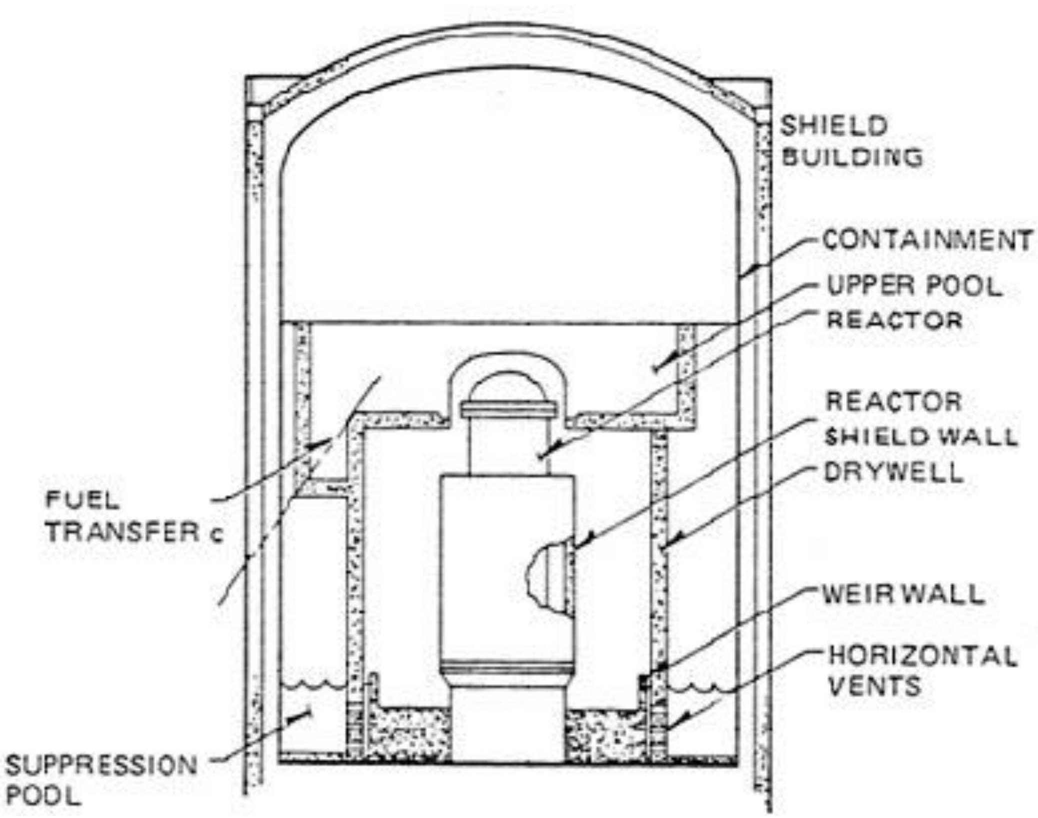


Figure 13. Steel shell and concrete BWR containment structure showing the pressure suppression pool.

3.4 CORE VESSEL

Table 2 gives the operational characteristics of this design.

Table 2. Operational Characteristics of a typical BWR.

Characteristic	Value
Thermal Power output	3,579 MWth
System pressure	1,040 psia
Fuel enrichment	2.2-2.7 %
Coolant flow	1.05×10^8 lbs/hr
Core inlet enthalpy	527.9 Btu/lb
Average exit quality	14.6 %
Core average void fraction	42.6 %
Maximum exit void fraction	76 %
Average inlet velocity	7.1 ft/sec
Core pressure drop	25.9 psi
Inlet temperature, feed water	420 ^oF
Core inlet temperature	532 ^oF
Outlet temperature	547 ^oF
Maximum fuel temperature	3,500 ^oF
Average linear heat rate	6.0 kW/ft
Maximum linear heat rate	13.4 kW/ft
Average heat flux	159,000 BTU/(hr.ft^2)
Maximum heat flux	354,000 BTU/(hr.ft^2)
Minimum CHFR	>1.9
Active height	148 inches
Equivalent active diameter	144 inches
Height to diameter ratio	1.03
Active core volume	2260 ft^3
Average core power density	1580 kW/ft^3
Fuel weight	138,000 kgs
Specific power	25.9 kW/kg U
Burnup	27,500 MW.days/MTU
Conversion ratio	0.5
Number of fuel assemblies	732
Fuel element array	8x8
Assembly dimensions	5.52 in x 5.52 in
Assembly pitch	6.0 in
Number of fuel rods per assembly	62
Total number of fuel rods	45,384
Fuel rod outer diameter	0.493 in
Fuel rod pitch	0.64 in
Pitch to diameter ratio	1.3
Cladding thickness	0.034 in
Fuel pellet diameter	0.416 in
Pellet to clad gap size	0.0045 in
Pellet density, percent of theoretical	94 %
Fission gas plenum length	12 in

3.5 FUEL ASSEMBLIES

The nuclear fuel assemblies are positioned inside a core shroud within the reactor vessel. Subcooled water is introduced at the bottom of the core and flows upward through vertical fuel assemblies.

The fuel rods are arranged in core with a cross sectional area equivalent to a circle of 12 feet diameter. The core consists of 700 fuel assemblies with a square cross section 5.52 inches on the side. Groups of four fuel assemblies are arranged around the cruciform control rod.

Fuel rods are arranged in an 8 x 8 array. They contain 62 active fuel rods and 2 hollow tie rods which are supported by the upper and lower tie plates. Each fuel assembly is surrounded by a Zircaloy-4 channel. The presence of this channel prevents the flow between adjacent fuel assemblies. The flow to individual fuel assemblies can thus be equipped with orifices so as to maintain a uniform exit quality from all fuel assemblies.

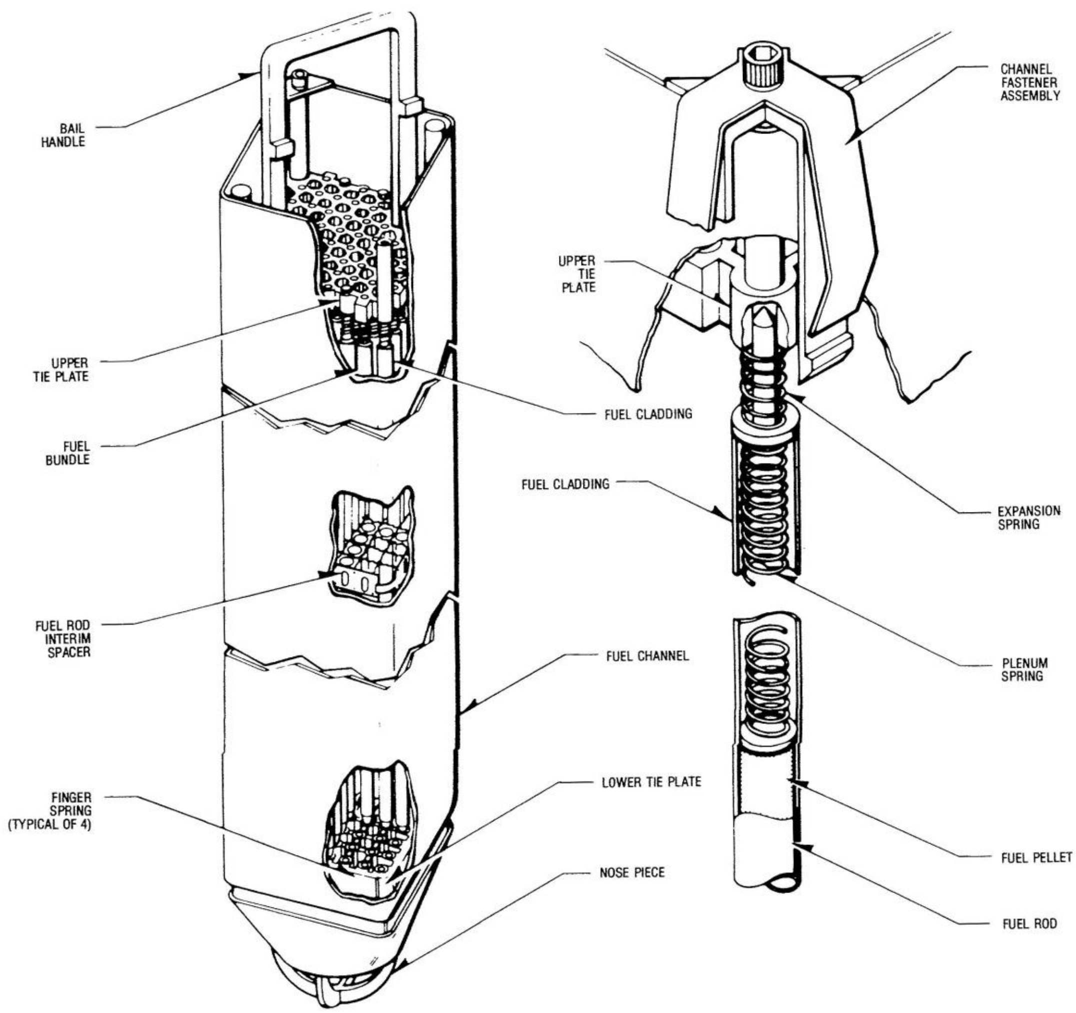


Figure 14. BWR fuel assembly.

The fuel consists of UO_2 pellets with an enrichment varying from 2.2 to 2.7 percent in U^{235}. The fuel pellets are 0.416 in in diameter and 0.41 in in length. The pellets are contained in in Zircaloy-2 tube with a wall thickness of 0.034 in. The fuel pellets are at a clearance from the cladding of 0.0045 in.

A two phase mixture at about 14 percent quality exits from the fuel assemblies and enters the upper plenum region in a domed top. An array of standpipes is attached to the dome with a three stage steam separator at the top of each standpipe. The centrifugal force separates the water from the mixture in each stage.

3.6 JET PUMPS

The water in the pool surrounding the standpipes is mixed with feed water by means of the feed water sparger. The water then flows to the down-comer annulus where the coolant circulation inside the core is achieved by the use of a set of 20 jet pumps.

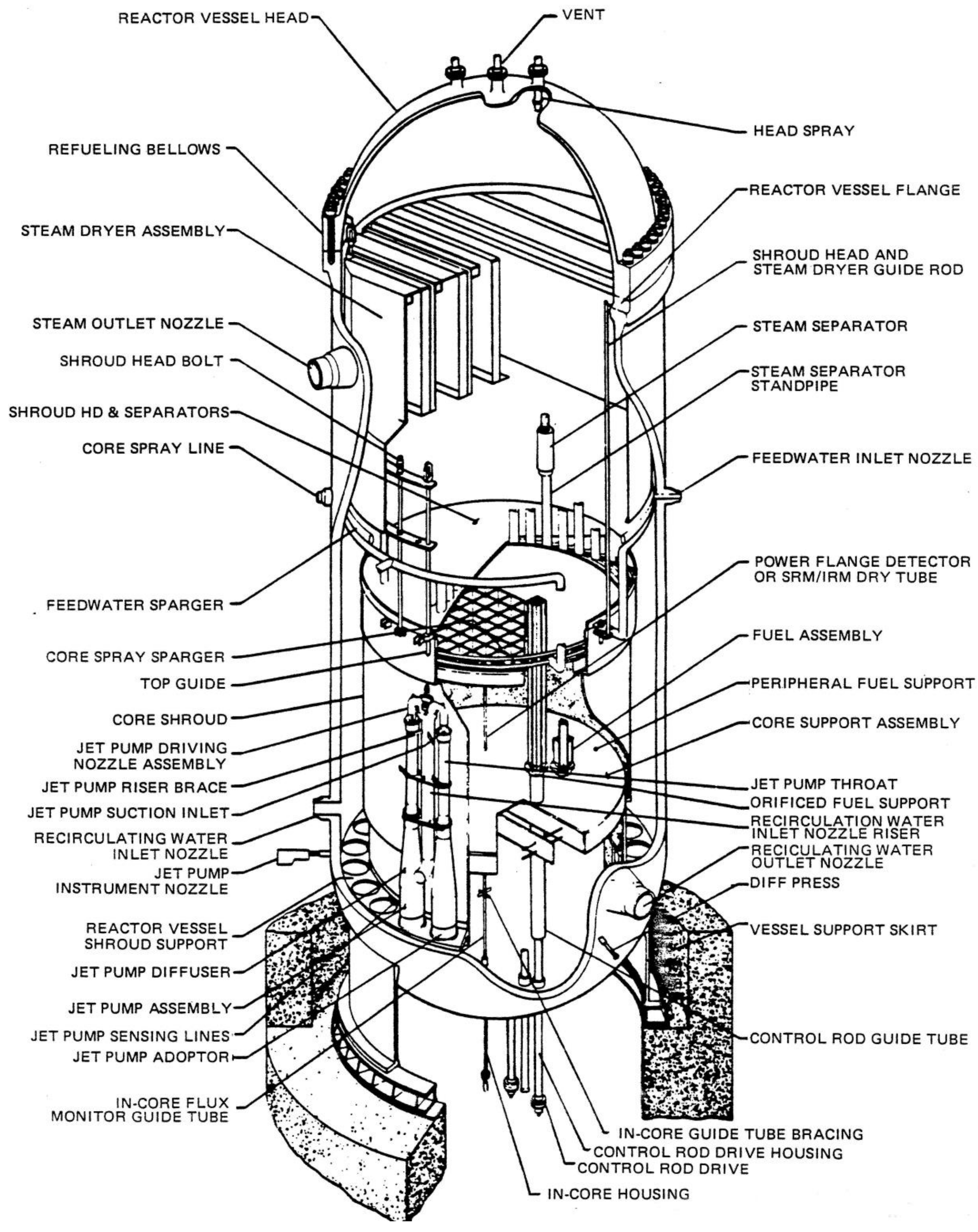


Figure 15. BWR Pressure vessel and internal components.

The jet pumps are mounted internally inside the reactor core and have no moving parts inside the reactor vessel. Water flow is induced through the Bernoulli effect generating a suction pressure inside the jet pump nozzle with a secondary flow by the coolant pumps positioned outside the reactor vessel.

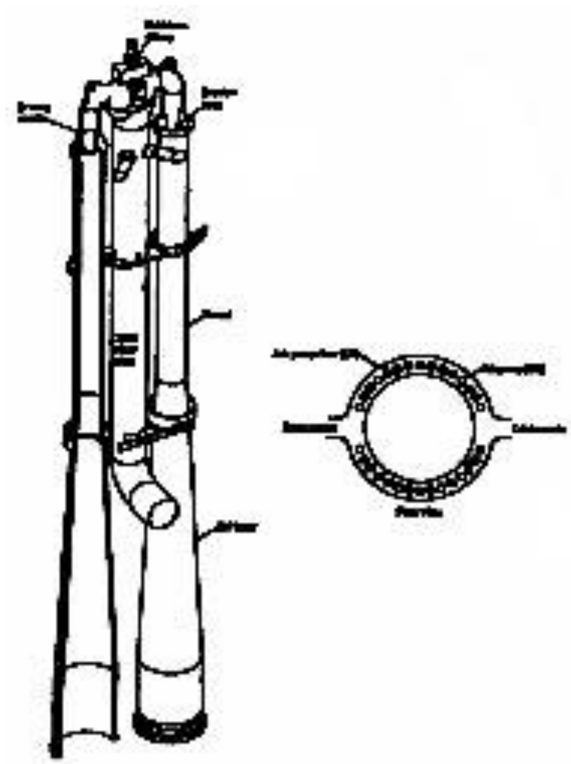


Figure 16. BWR jet pumps.

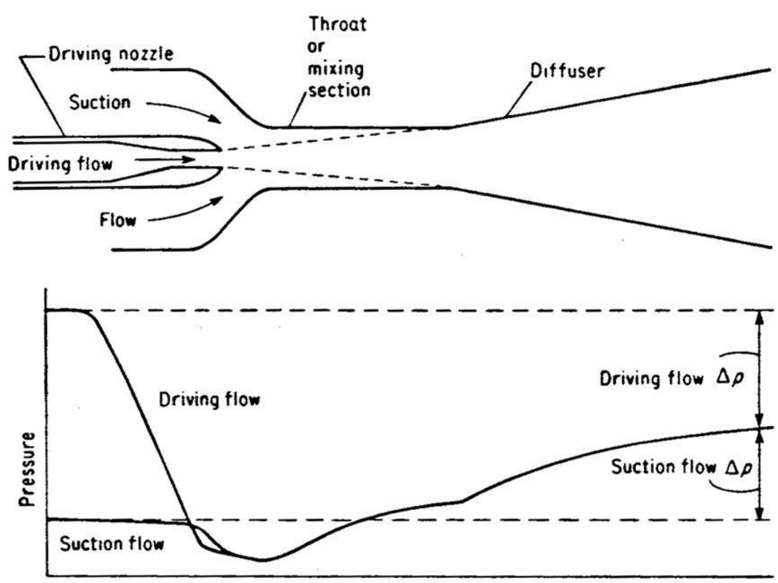


Figure 17. The Bernoulli Effect generates suction in BWR jet pump, without moving parts.

3.7 FLOW SYSTEM

The coolant recirculation system consists of two loops external to the reactor vessel. One third of the recirculation water enters the recirculation pumps. The recirculation water prfessure is increased by the pumps and is distributed to a manifold from which connecting pipes are welded to the recirculation water inlet nozzles of the rector vessel. One inlet nozzle serves two jet pumps

Since the BWR design does not have a separate steam generator like the PWR design, water chemistry becomes paramount to avoid corrosion and activation of the corrosion products. These activated products can accumulate at hot spots in the reactor circuitry leading to a radiation dose to the operating personnel.

The flow diagram for the Quad Cities BWR is shown in Fig. 19. It shows the cleanup demineralizer system. A moisture separator is used between the high pressure and low pressure turbines. The condensate storage tank is used to operate the control rod drives and serves as a supply of cooling water in the case of an emergency.

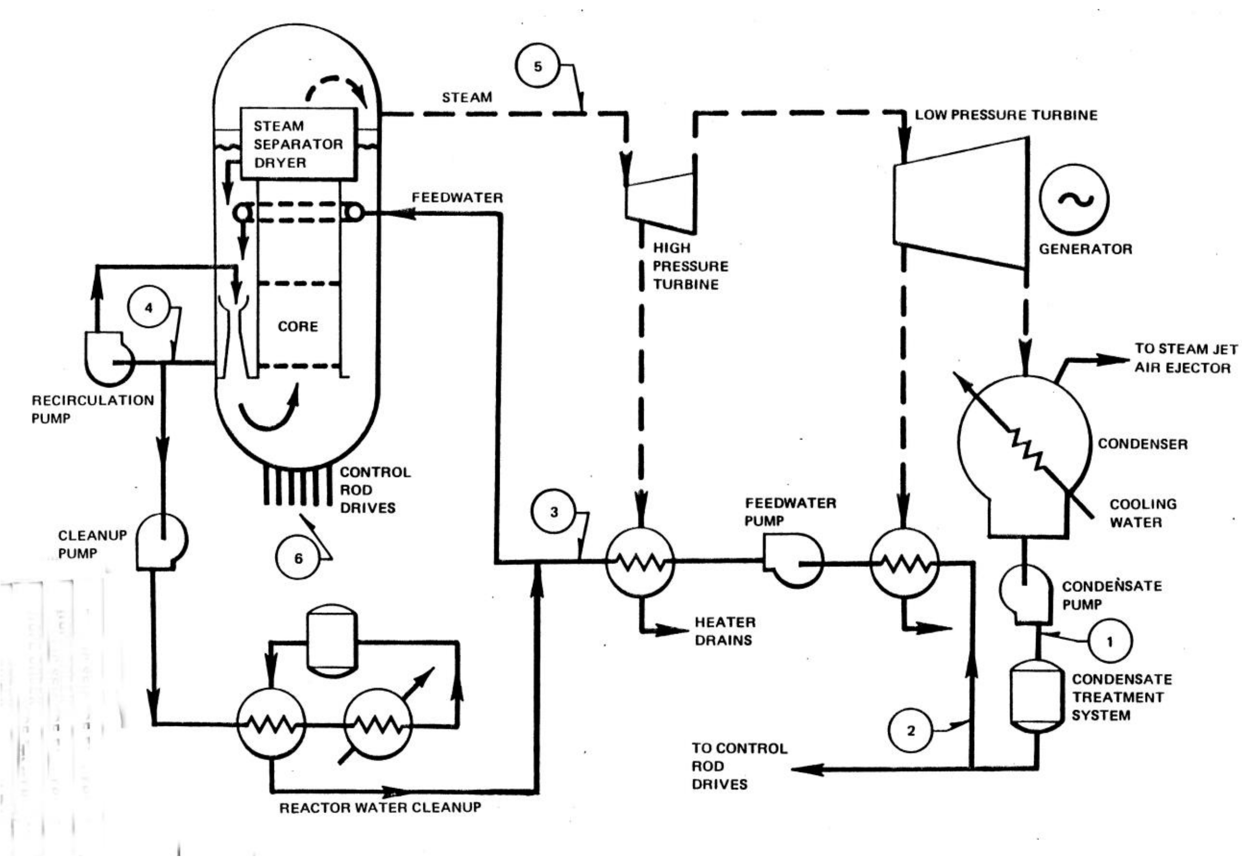


Figure 18. Typical BWR flow diagram.

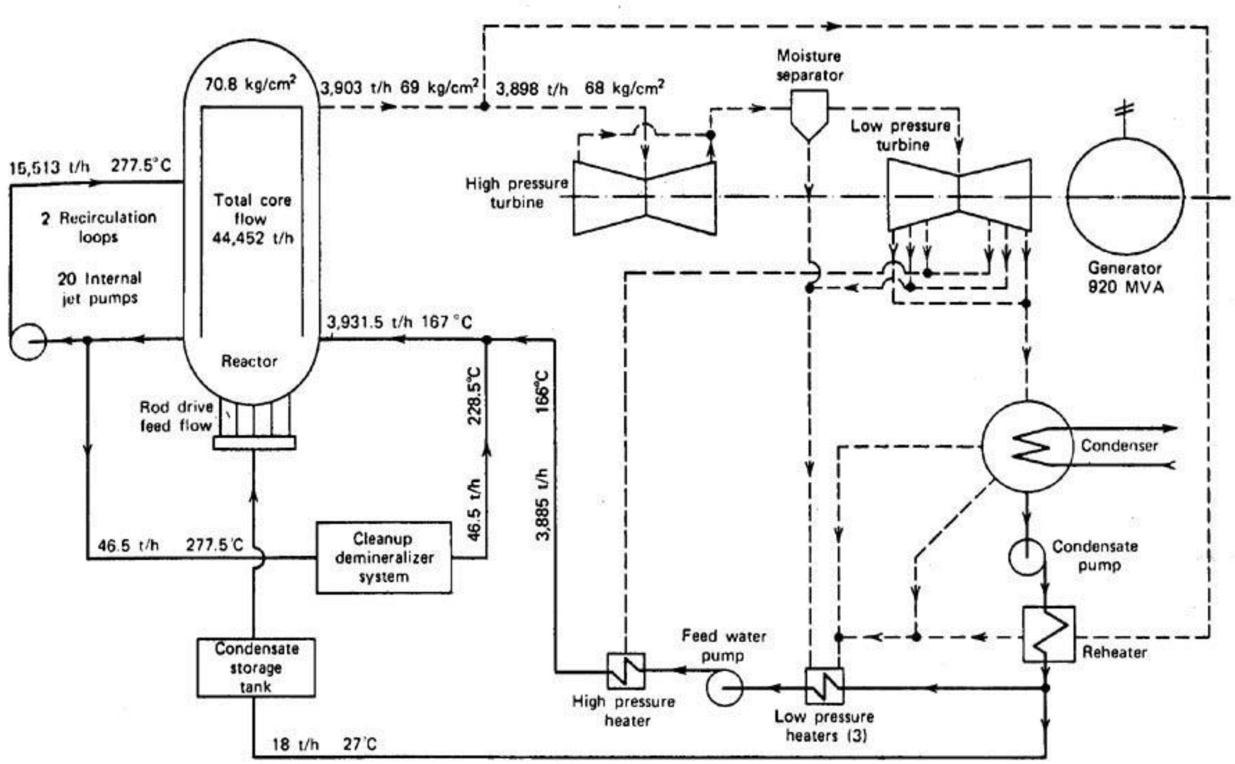
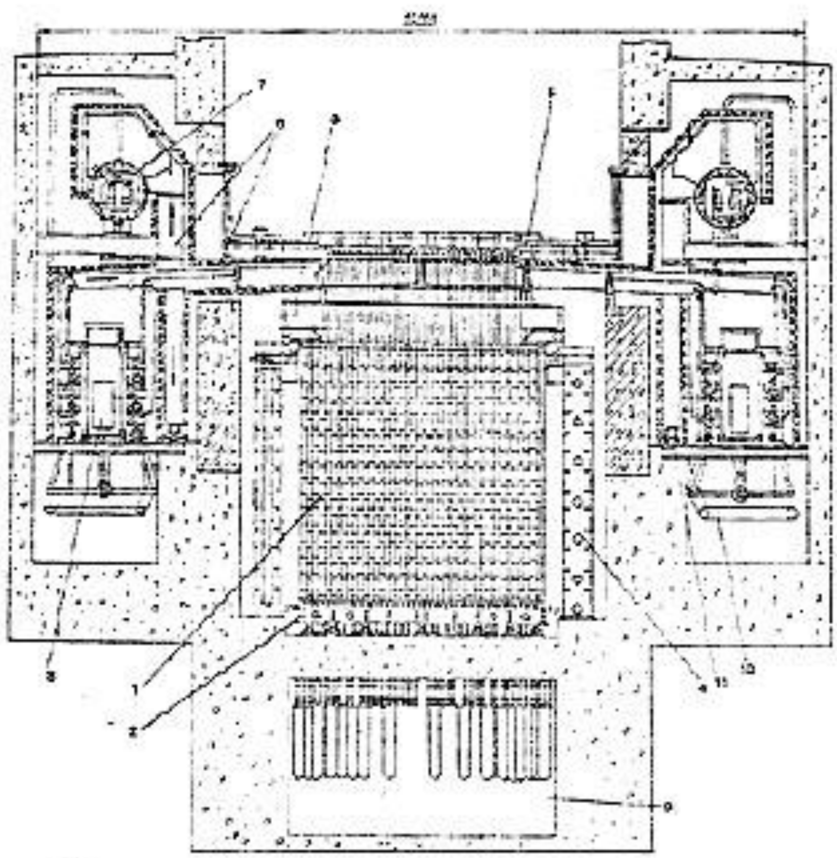


Figure 19. Quad-Cities BWR flow diagram.

3.8 BOILING TUBE REACTOR DESIGN

The boiling tube reactor design is used in the RBMK-1000 Russian design. It uses light water as the boiling medium and graphite as a moderator. The steam is separated from the two phase mixture in steam drums.

The multiple tubes design offers a level of safety since in the case of coolant leakage in a tube it could be isolated from the rest of the system. However the use of light water as a coolant, and its acting as a neutron poison leads to a positive power coefficient of reactivity. When the temperature increases leading to water boiling, this corresponds to a positive reactivity insertion, and consequent power level rise. This positive feedeback effect, can make the system unstable at low power level. This type of behavior contributed to the Chernobyl accident which occurred in this type of design.



Beloyarsk BWR Unit No. 2: Vertical Cross section through reactor. 1–Graphite stack, 2–Lower plate, 3–Upper plate, 4–Water shielding, 5–Upper floor, 6–Piping, 7–Steam separator, 8–Service corridor, 9–Control-rod drive compartment, 10–Piping, and 11–Superheat steam header.

Figure 20. RBMK-1000 Boiling Light Water, BLW graphite moderated tube reactor. Steam is separated from two phase mixture in steam drums.

Figure 21. Top view of RBMK-1000 core during on line refueling.

3.9 ADVANCED BWR, ABWR

The Advance Boiling Water Reactor (ABWR) is an evolutionary design providing a higher chimney and consequently better natural convection cooling in the core. The Advanced Boiling Water Reactor (ABWR) already has track record. In Japan, four ABWR units are in operation; another three units are under construction in Taiwan and Japan, and nine more units are planned in Japan. It is feasible that an ABWR plant could be built in the USA and be commercially operational by 2012.

The design is available today for immediate generation needs of about 1500 MWe, providing technology and schedule certainty, along with competitive economics.

The ABWR is a direct cycle BWR that reflects 50 years of continued evolution from GE's initial BWR concept. It combines the best features of GE's worldwide BWR fleet with advanced technology enhancements, such as digital controls, that improve performance and longevity. The ABWR design is already licensed in three countries: the USA, Japan and Taiwan.

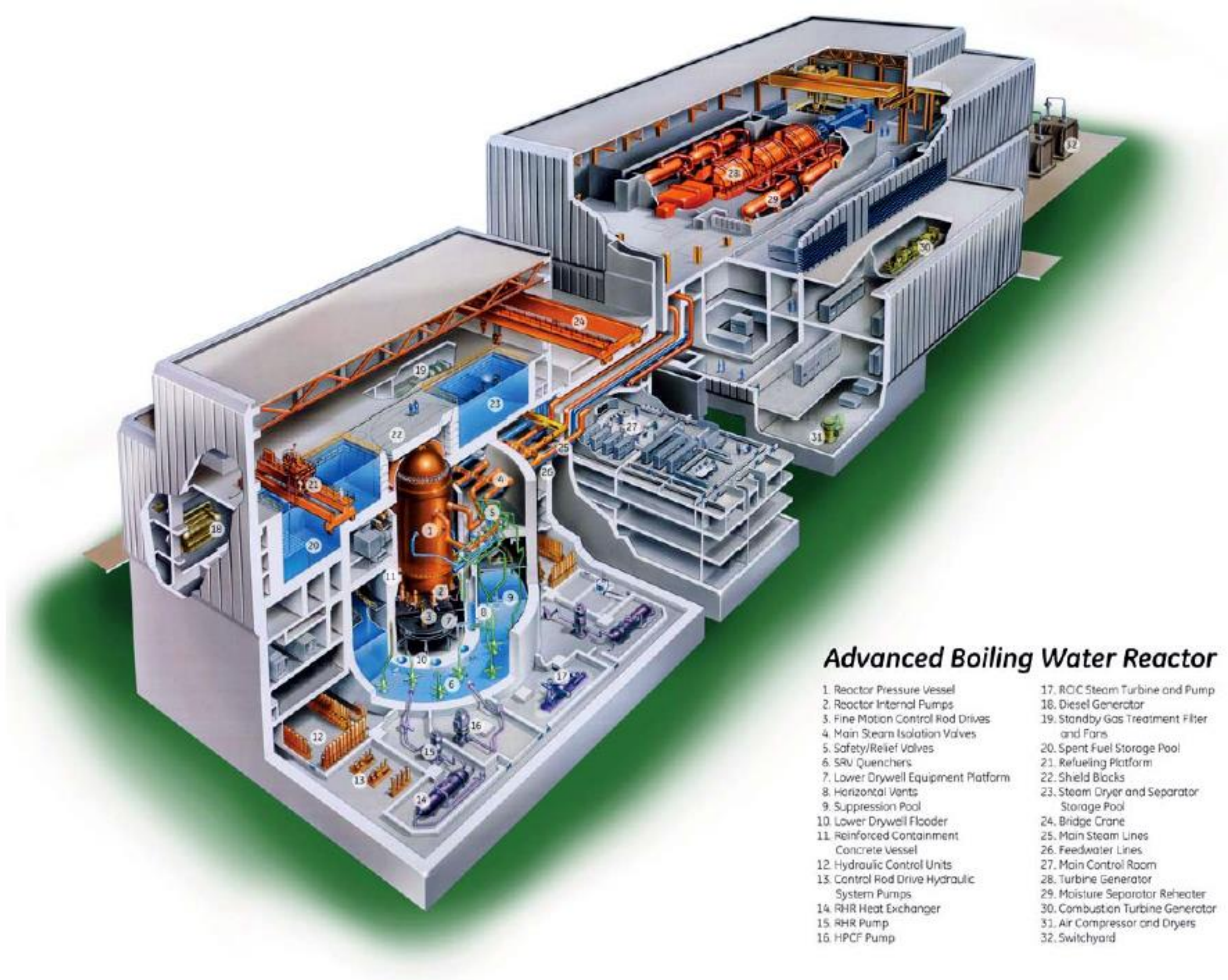


Figure 22. Advanced Boiling Water Reactor, ABWR Plant Design. Source: GE.

Figure 23. Pressure vessel of the Advanced Boiling Water Reactor, ABWR. Source: GE.

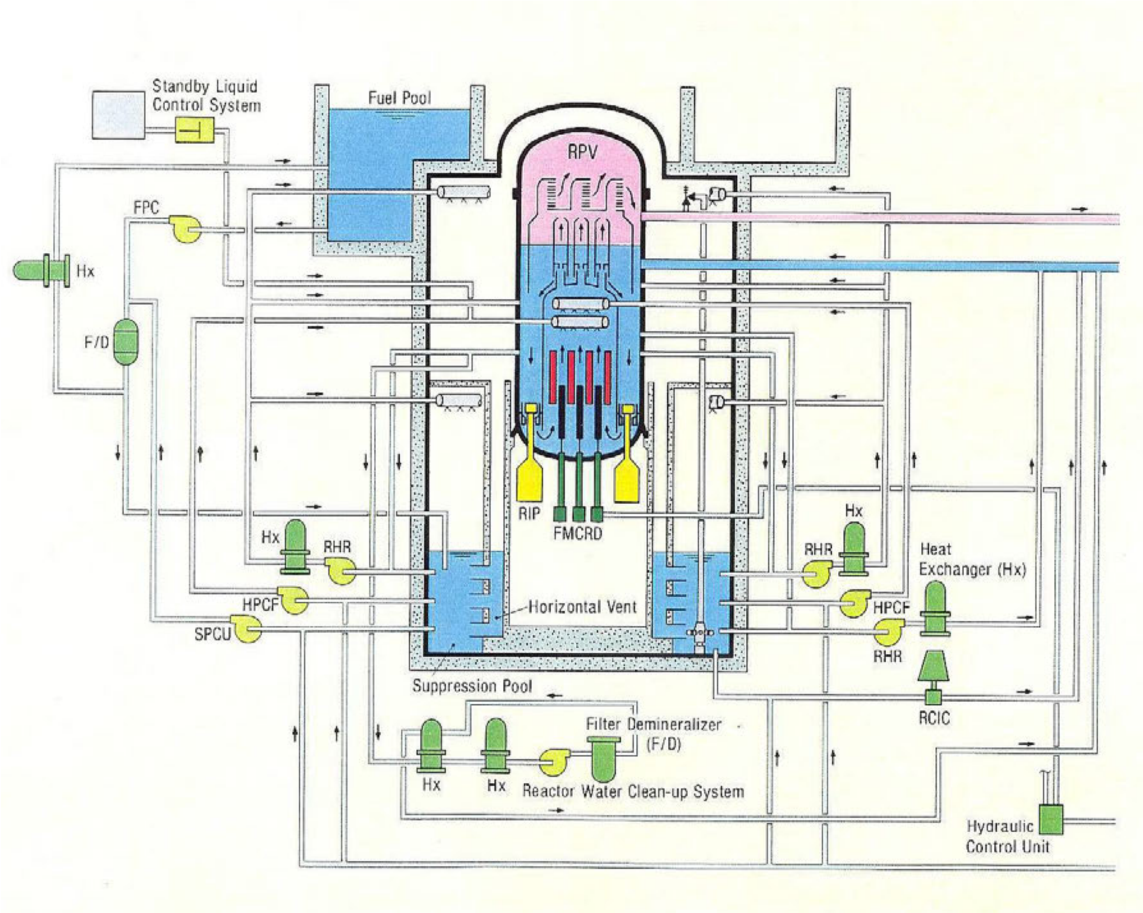


Figure 24. Flow Diagram of Advanced Boiling Water Reactor (ABWR). Source: Tepco.

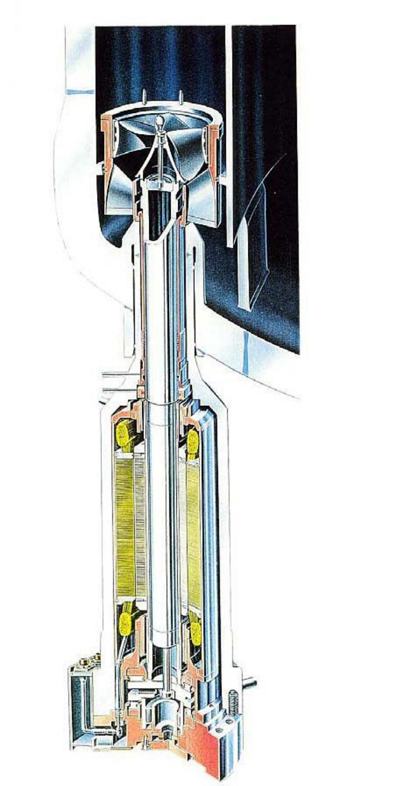


Figure 25. Canned Rotor external coolant pump for the Advanced Boiling Water Reactor (ABWR). Source: Tepco.

3.10 ECONOMICAL SMALL BOILING WATER REACTOR, ESBWR

OVERVIEW

The next evolution of advanced BWR technology is the ESBWR. It utilizes a number of new features to provide better plant security; improved safety; more location options; excellent economics; and operational flexibility that ultimately increases plant availability.

In 1992, the GE Company began to design a natural circulation, boiling water reactor featuring passive safety systems. This effort initially produced a 670 MWe reactor known as the Simplified BWR, SBWR.

The development program was later redirected to design a larger reactor that used economies of scale, proven technology, and components from the Advanced BWR, ABWR to create a new reactor at reduced capital cost. The new advanced reactor is known as the Economic Simplified BWR, ESBWR.

Some of the ESBWR primary features are:

1. Simplified design features: Passively removes decay heat directly to the atmosphere. Eleven systems are eliminated from the previous designs. It involves the use of 25 percent fewer pumps, valves and motors than other BWR designs.

2. Passive design features reduce the number of active systems, increasing safety: It is 11 times more likely for the largest asteroid near the earth to impact the earth over the next 100 years than for an ESBWR operational event to result in the release of fission products to the environment.
3. Incorporation of features used in other operationally-proven reactors, including passive containment cooling, isolation condensers, natural circulation and debris resistant fuel.
4. Natural circulation and convection cooling is used.
5. Expedited construction schedule because of standardized modules and pre-licensed design. Estimated construction times for first concrete to first core load of 36 months

Table 3. Technical Specifications of the ESBWR.

Parameter	Value
Power plant	
Electrical power	1,560 MWe
Thermal output	4,500 MWth
Estimated cost	$1160-1250 /kWe
Fuel type	GE14, 10x10 fuel assembly
Design life	60 years
Thermal efficiency	34.7 percent
Core	
Fuel type	Enriched UO_2
Enrichment	4.2 percent
Number of fuel bundles	1,132
Coolant	Light water
Moderator	Light water
Operating cycle length, operation time between refuelings	12-24 months
Outage duration, for refueling only, 24 months cycle	14 days
Percent fuel replaced at refueling	20 percent for a 12 months cycle 42 percent for a 24 months cycle
Average fuel burnup at discharge	50,000 MWd/mt
Number of steam lines	4
Number of feedwater trains	2
Containment parameters	
Design temperature	340 ^oF
Design pressure	45 psig
Reactor parameters	
Design temperature	575 ^oF
Operating temperature	550 ^oF
Design pressure	1,250 ^oF
Nominal operating pressure	1,040 psia
Feedwater and turbine parameters	
Turbine inlet / outlet temperature	543 / 93 ^oF

Turbine inlet / outlet pressure	985 / 0.8 psia
Feedwater temperature	420 ^{o}F
Feddwater pressure	1,050 psia
Feedwater flow	4.55 x 10^4 gpm
Steam mass flow rate	19.31 x 10^6 lbs / hr
Waste generation	
High level, spent fuel	50 mt / year
Intermediate level (spent resins, filters) and low level (compactable / noncompactable)	1,760 ft^3 /year

DESCRIPTION

The ESBWR plant design relies on the use of natural circulation and passive safety features to enhance the plant performance and simplify the design. The use of natural circulation allows the elimination of several systems.

The ESBWR utilizes the isolation condenser system for high pressure inventory control and decay heat removal. After initiation of the automatic depressurization system, low pressure inventory control is provided by the gravity driven cooling system.

Containment cooling is provided by the Passive Containment Cooling System, PCCS. The primary intent of the ESBWR is to significantly reduce both capital and Operation and Maintenance, O&M costs of the earlier SBWR and ABWR plants.

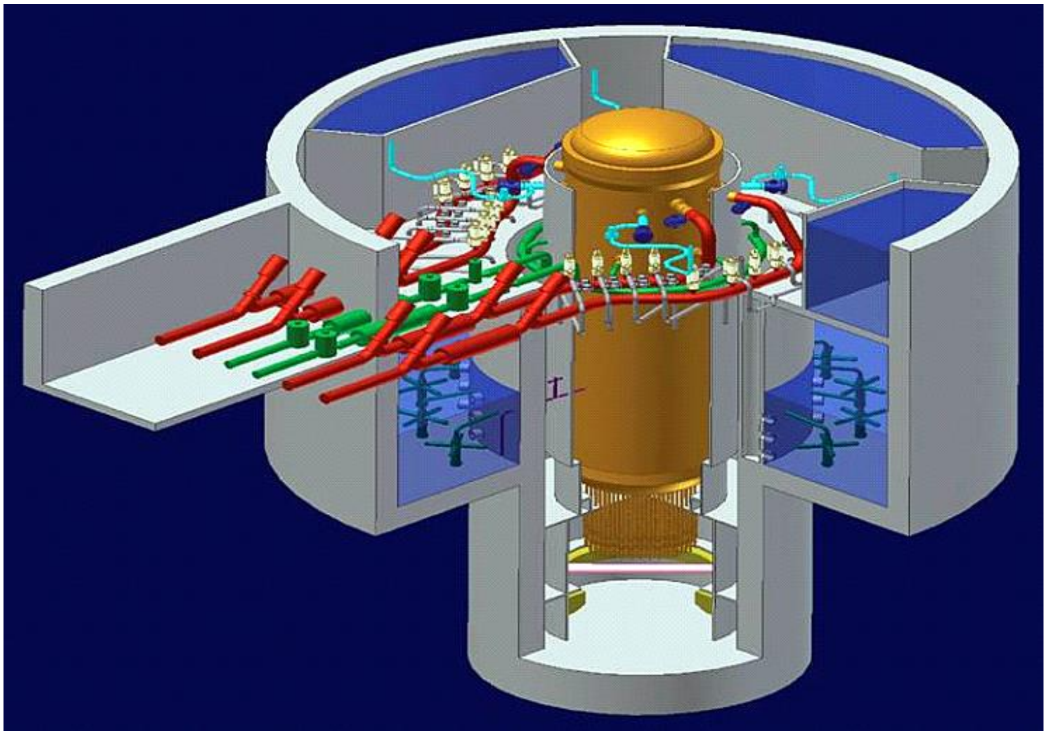


Figure 26. ESBWR cutout. Source: GE.

CONSTRUCTION

The ESBWR will rely on ABWR construction techniques, primarily the use of open top modular construction. A construction schedule of 45 months from first concrete to commercial operation is anticipated for the first-of-a-kind unit.

There is high confidence in the design because it uses standard, proven equipment including extensive use of ABWR components and fully tested passive safety features from the SBWR.

Testing was performed to verify natural circulation and the use of several components in new applications.

This design relies heavily on the same basic components as the ABWR, which has been installed previously. GE has been involved in the construction of 64 BWRs. There are currently 54 GE BWRs operating worldwide.

The ESBWR has achieved its basic plant simplification by incorporating innovative adaptations of operating plant systems into the plant design. This includes combining shutdown cooling and reactor water cleanup systems. The only major new system is the PCCS. The reactor building is reduced in volume. Nearly all safety systems are now located in the containment structure or directly above it. This allows significant reductions in the volume and footprint of other buildings.

Figure 27. ESBWR pressure vessel. Natural circulation is established because of density differences between water in the vessel annulus and the steam/water mixture inside the shroud

and chimney. Natural circulation is enhanced by the shorter fuel, 8.6 meter chimney, improved steam separator, and opening the flow path between the downcomer and the lower plenum. Source: GE.

SAFETY SYSTEMS OF THE ESBWR

The safety systems in the ESBWR are passive and include:

1. Automatic Depressurization System, ADS

The ADS consists of:
i. Ten Safety Relief Valves, SRVs mounted on top of the main steam lines that discharge steam to the suppression pool
ii. Eight Depressurization Valves, DPVs that discharge steam to the drywell.

2. Gravity Driven Cooling System, GDCS

The makeup water gravity flows into the vessel after the ADS depressurizes the reactor vessel. The GDCS pool capacity is primarily determined by containment geometrical considerations.

The GDCS and ADS constitute the plant's Emergency Core Cooling System (ECCS).

3. Isolation Condenser System, ICS

The ICS removes decay heat from the reactor following transient events involving reactor scram including station blackout. The ICS consists of four independent high pressure loops, each containing a heat exchanger that condenses steam on the tube side. The tubes are in a large pool, outside the containment. The system uses natural circulation to remove decay heat.

4. Passive Containment Cooling System, PCCS

The PCCS removes heat from inside containment following a loss of-coolant accident (LOCA). The system consists of four safety-related low-pressure loops. Each loop has a heat exchanger open to the containment, a condensate drain line and a vent discharge line submerged in the suppression pool. The four heat exchangers, similar in design to the isolation condensers, are located in cooling pools external to the containment. The PCCS limits containment pressure to < 40 psig.

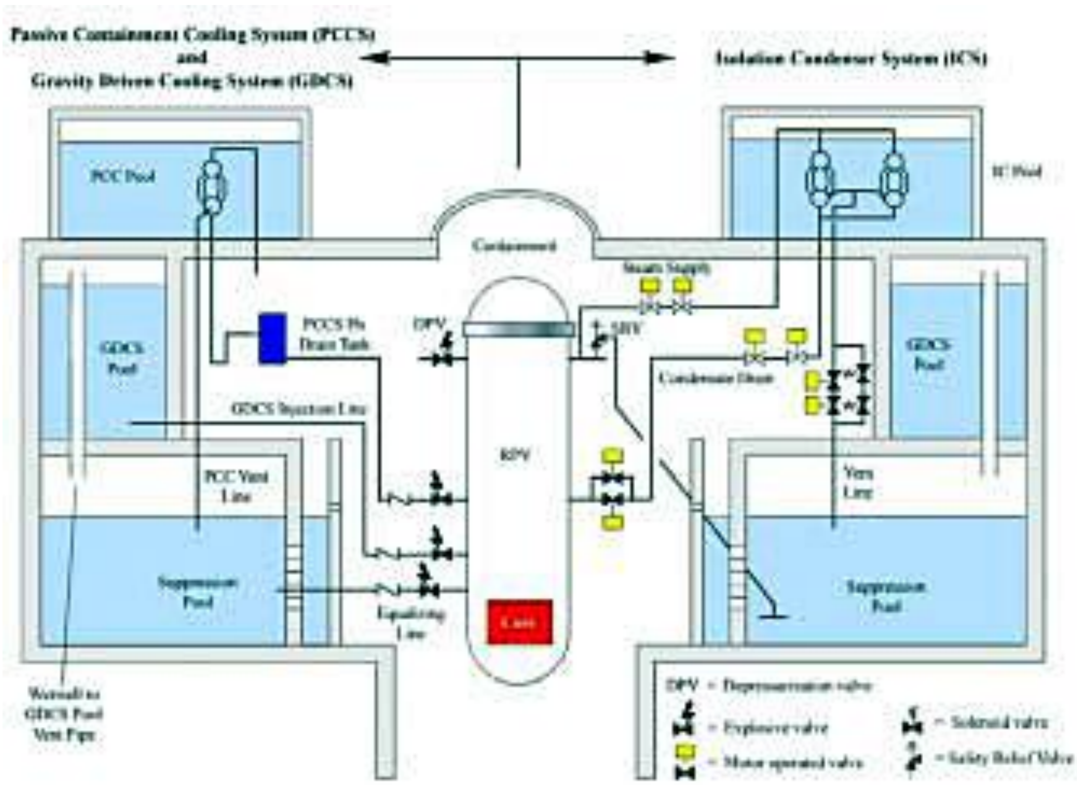


Figure 28. Safety systems in the ESBWR.

3.11 ENGINEERED SAFETY FEATURES

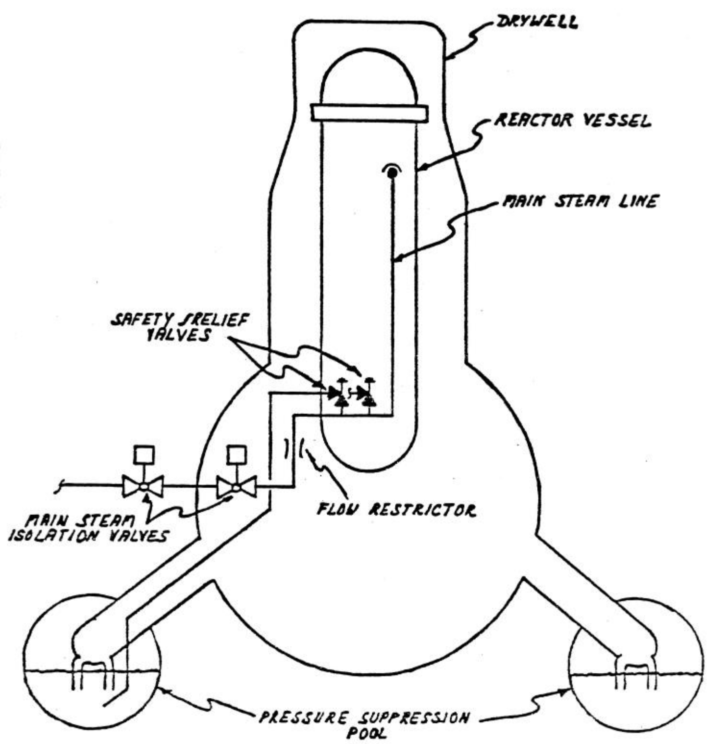


Figure 29. BWR depressurization System.

Figure 30. BWR main steam isolation valves.

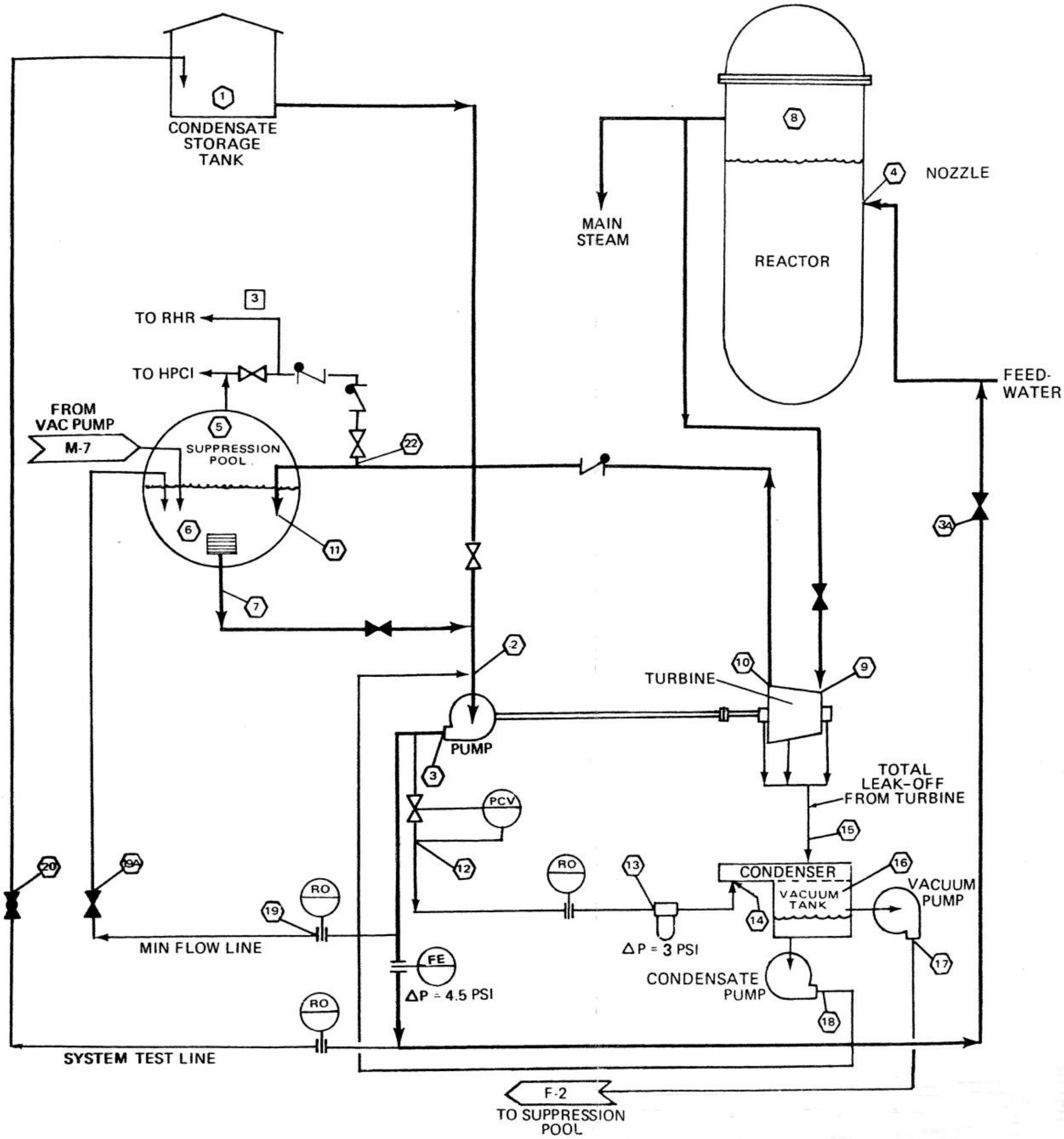


Figure 31. Reactor Core Isolation Cooling System, RCICS.

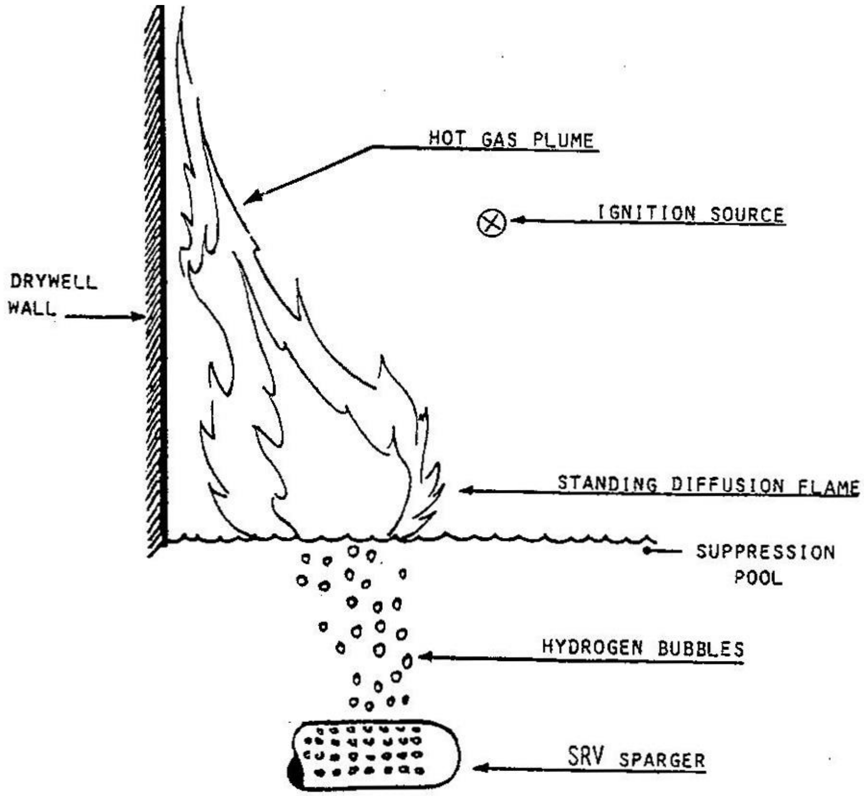


Figure 32. Hydrogen sparging system for controlled hydrogen burning.

Figure 33. Emergency Battery Banks for backup power.

3.12 COOLANT ACTIVATION AND N^{16} FORMATION

INTRODUCTION

In light water and heavy water moderated and cooled reactors, the threshold fast neutron activation set of reactions with the isotopes of oxygen in the water:

$$\begin{aligned} &{}_8O^{16} + {}_0n^1 \rightarrow {}_1H^1 + {}_7N^{16} \\ &{}_7N^{16} \rightarrow {}_8O^{16} + {}_{-1}e^0 + \gamma \end{aligned} \tag{1}$$

with: $T_{1/2}\left({}_7N^{16}\right) = 7.1 \sec$,

E_γ = 6.13 MeV in 69 percent of the decays,

$\sigma_{activation}$ = 46 mb,

Natural abundance of $_8O^{16}$ = 99.758 percent,

$\rho(H_2O) = 1.0\ [gm/cm^3]$

and the set of reactions:

$$\begin{aligned} &{}_8O^{17} + {}_0n^1 \rightarrow {}_1H^1 + {}_7N^{17} \\ &{}_7N^{17} \rightarrow {}_8O^{17*} + {}_{-1}e^0 + \gamma \\ &{}_8O^{17*} \rightarrow {}_8O^{16} + {}_0n^1 \end{aligned} \tag{2}$$

with: $T_{1/2}\left({}_7N^{17}\right) = 4.17 \sec$,

$\sigma_{activation}$ = 5.9 mb,

Natural abundance of $_8O^{17}$ = 0.038 percent,

* denotes an excited state,

are significant, particularly in the Boiling Water Reactor (BWR), because of the short transit time of the generated steam between the reactor core and the turbine and other equipment external to the reactor shield.

RATE EQUATIONS

The rate equations describing the production and decay of N^{16} and N^{17} are respectively:

$$\begin{aligned} \frac{dN^{16}}{dt} &= R({}_8O^{16}) - \lambda^{N^{16}} N^{16} \\ \frac{dN^{17}}{dt} &= R({}_8O^{17}) - \lambda^{N^{17}} N^{17} \end{aligned} \tag{3}$$

or:

$$\frac{dN_i}{dt} = R_i - \lambda_i N_i, i = 1,2. \tag{4}$$

SOLUTION OF THE RATE EQUATIONS FOR ASINGLE LOOP THROUGH THE REACTOR

For a single loop through the reactor, we solve the rate equations accounting for the production and decay rates of the $_7N^{16}$ and $_7N^{17}$ in nuclei per unit volume, assuming their initial concentrations at time t=0 to be negligible.

By multiplying both sides of Eqn. 4 by an integrating factor:

$$e^{\lambda_i t}$$

we get:

$$e^{\lambda_i t}\frac{dN_i}{dt} = e^{\lambda_i t}R_i - e^{\lambda_i t}\lambda_i N_i$$

$$e^{\lambda_i t}\frac{dN_i}{dt} + \lambda_i N_i . e^{\lambda_i t} = R_i . e^{\lambda_i t}$$

The left hand side can be written as a total differential as:

$$\frac{d}{dt}(N_i . e^{\lambda_i t}) = R_i . e^{\lambda_i t} \tag{5}$$

which can be proved by applying the chain rule of differentiation.

Separating the variable and integrating using limit integration, we get:

$$\int_{N_{i0}}^{N_i(t)} d(N_i . e^{\lambda_i t}) = R_i . \int_0^t e^{\lambda_i t} dt$$

$$N_i(t)e^{\lambda_i t} - N_{io}.1 = \frac{R_i}{\lambda_i}(e^{\lambda_i t} - 1)$$

Multiplying both sides by:

$$e^{-\lambda_i t}$$

$$N_i(t) = N_{io}.e^{-\lambda_i t} + \frac{R_i}{\lambda_i}(1 - e^{-\lambda_i t}) \tag{6}$$

For negligible nuclei at time t = 0, we get:

$$N_i(t) = \frac{R_i}{\lambda_i}(1 - e^{-\lambda_i t}) \quad (7)$$

The activities of the species can be written as:

$$A_i(t) = \lambda_i N_i(t) = R_i(1 - e^{-\lambda_i t}), i = 1, 2 \quad (8)$$

Writing Eqn. 8 explicitly yields for the activities after a single loop through the reactor:

$$\begin{aligned} A_{16}(t) &= \lambda_{16} N_{N^{16}}(t) = R_{16}(1 - e^{-\lambda_{16} t}) \\ A_{17}(t) &= \lambda_{17} N_{N^{17}}(t) = R_{17}(1 - e^{-\lambda_{16} t}) \end{aligned} \quad (9)$$

PRODUCTION RATES

The rates of production of N^{16} and N^{17} per unit volume can be calculated from:

$$\begin{aligned} R_i &= \Sigma_{activation,i}\phi \\ &= N_i \sigma_{activation,i}\phi \end{aligned} \quad (10)$$

where:
ϕ is the average neutron flux [$\frac{\text{n}}{\text{cm}^2.\text{sec}}$]
$\sigma_{activation,i}$ is the microscopic activation cross section [b]
$\Sigma_{activation,i}$ is the macroscopic activation cross section [cm^{-1}]

Using the modified form of Avogadro's law:

$$N_i = a_i \frac{\rho A_v}{M} \quad (11)$$

where:
a_i are the percentage natural abundances
A_v is Avogadro's number
ρ is the density of water [$\frac{\text{gm}}{\text{cm}^3}$]
M is the molecular weight of water [amu]

Substituting Eqn. 11 into Eqn. 10 yields:

$$R_i = a_i \frac{\rho A_v}{M} \sigma_{activation,i}\phi \quad (12)$$

For the data of N^{16} and N^{17} and a neutron flux of $\phi = 10^{10}[\frac{n}{cm^2.\text{sec}}]$:

$$R_{16} = \frac{99.758}{100}\frac{1\times 0.6\times 10^{24}}{18} 46\times 10^{-3}\times 10^{-24}\times 10^{10} = 1.53\times 10^{7}[\frac{\text{nuclei}}{\text{cm}^3.\text{sec}}]$$
$$R_{17} = \frac{0.038}{100}\frac{1\times 0.6\times 10^{24}}{18} 5.9\times 10^{-3}\times 10^{-24}\times 10^{10} = 7.47\times 10^{2}[\frac{\text{nuclei}}{\text{cm}^3.\text{sec}}]$$

It can be readily noticed that the production rate of N^{16} exceeds that of N^{17}:

$$R_{16} > R_{17}$$
$$\frac{R_{16}}{R_{17}} = \frac{1.53\times 10^{7}}{7.47\times 10^{2}} = 2.05\times 10^{4}$$

ONE PATH THROUGH CORE AND OUTSIDE LOOP

If the transit time through the reactor is t_c, the activity after one loop in the reactor loop would be from Eqn. 8:

$$A_i(t) = \lambda_i N_i(t) = R_i(1-e^{-\lambda_i t_c}) \tag{13}$$

After one loop in the reactor core and a time t_o in the outside loop, the total transit time is:

$$T = t_c + t_0 \tag{14}$$

The activity acquired after one full path through the reactor core and outside loop, where it decays is becomes:

$$A_{i1} = R_i(1-e^{-\lambda_i t_c})e^{-\lambda_i t_o} \tag{15}$$

MULTIPLE PATHS ACTIVATION

After two paths through the core and the outside loop the activities would be from Eqn. 15:

$$\begin{aligned} A_{i2} &= R_i(1-e^{-\lambda_i t_c})e^{-\lambda_i t_0}e^{-\lambda_i T} + R_i(1-e^{-\lambda_i t_c})e^{-\lambda_i t_0} \\ &= R_i(1-e^{-\lambda_i t_c})e^{-\lambda_i (t_o+T)} + R_i(1-e^{-\lambda_i t_c})e^{-\lambda_i t_0} \\ &= R_i(1-e^{-\lambda_i t_c})e^{-\lambda_i t_0}(1+e^{-\lambda_i T}) \end{aligned} \tag{16}$$

After n loops Eqn. 16 can be generalized as:

$$A_{in}(t) = R_i(1-e^{-\lambda_i t_c})e^{-\lambda_i t_0}(1+e^{-\lambda_i T}+...+e^{-(n-1)\lambda_i T}) \quad (17)$$

Since:

$$\sum_{i=0}^{n} e^{-ix} = \frac{1-e^{-nx}}{1-e^{-x}} = (1+e^{-x}+...+e^{-nx}) \quad (18)$$

With $x \equiv \lambda_i T$ it becomes:

$$\sum_{i=0}^{n-1} e^{-i\lambda_i T} = \frac{1-e^{-(n-1)\lambda_i T}}{1-e^{-\lambda_i T}} = (1+e^{-\lambda_i T}+...+e^{-(n-1)\lambda_i T}) \quad (18)'$$

Then we can express Eqn. 17 for the activation after n loops as:

$$A_{in} = R_i(1-e^{-\lambda_i t_c})e^{-\lambda_i t_0}\frac{1-e^{-(n-1)\lambda_i T}}{1-e^{-\lambda_i T}} \quad (19)$$

EQUILIBRIUM ACTIVITIES

If the half-lives of the nuclides are not very long:

$$\text{(n-1)}\lambda_1\text{T} = \text{(n-1)}\frac{\ln 2}{T_{1/2,i}}\text{T} \gg 0,$$

and consequently as $n \to \infty$, Eqn. 19 reduces to:

$$\begin{aligned} A_{i\infty} &\approx R_i(1-e^{-\lambda_i t_c})e^{-\lambda_i t_0}\frac{1-0}{1-e^{-\lambda_i T}} \\ &\approx R_i\frac{1-e^{-\lambda_i t_c}}{1-e^{-\lambda_i T}}e^{-\lambda_i t_0} \end{aligned} \quad (20)$$

The equilibrium activities for N^{16} and N^{17} for:

$$\begin{aligned} t_c &\approx 2\,\text{sec}, \\ t_o &\approx 5\,\text{sec}, \\ T &\approx t_c + t_o = 2+5 = 7\,\text{sec}. \end{aligned}$$

become:

$$A_{16\infty} \approx 1.53\times10^7 \frac{1-e^{-\frac{\ln 2}{7.1}2}}{1-e^{-\frac{\ln 2}{7.1}7}} e^{-\frac{\ln 2}{7.1}5} = 1.53\times10^7 \frac{0.177}{0.495} 0.614 = 3.36\times10^6 [\frac{Bq}{cm^3}]$$

$$A_{17\infty} \approx 7.47\times10^2 \frac{1-e^{-\frac{\ln 2}{4.17}2}}{1-e^{-\frac{\ln 2}{4.17}7}} e^{-\frac{\ln 2}{4.17}5} = 7.47\times10^2 \frac{0.283}{0.688} 0.436 = 1.34\times10^2 [\frac{Bq}{cm^3}]$$

DECREASING THE EQUILIBRIUM ACTIVITIES

A possible design feature in Boiling Water Reactors (BWRs) is to divert the steam from the reactor to the top of the reactor building then to its bottom before feeding it to the turbine. This would increase the outside loop transit time to t_o = 10 seconds:

$$\begin{aligned} &t_c \approx 2\sec, \\ &t_o \approx 10\sec, \\ &T \approx t_c + t_o = 2 + 10 = 12\sec. \end{aligned}$$

In this case the equilibrium activities would become:

$$A_{16\infty} \approx 1.53\times10^7 \frac{1-e^{-\frac{\ln 2}{7.1}2}}{1-e^{-\frac{\ln 2}{7.1}12}} e^{-\frac{\ln 2}{7.1}10} = 1.53\times10^7 \frac{0.177}{0.690} 0.377 = 1.480\times10^6 [\frac{Bq}{cm^3}]$$

$$A_{17\infty} \approx 7.47\times10^2 \frac{1-e^{-\frac{\ln 2}{4.17}2}}{1-e^{-\frac{\ln 2}{4.17}12}} e^{-\frac{\ln 2}{4.17}10} = 7.47\times10^2 \frac{0.283}{0.864} 0.190 = 4.649\times10^1 [\frac{Bq}{cm^3}]$$

Thus doubling the outside loop transit time from 5 to 10 seconds, reduces the equilibrium activities for the two nitrogen isotopes by factors of:

$$N^{16} : \frac{3.360\times10^6}{1.480\times10^6} = 2.3$$

$$N^{17} : \frac{1.340\times10^2}{4.649\times10^1} = 2.9$$

REFERENCES

1. W. B. Cotrell, "The ECCS Rule-Making Hearing," Nuclear Safety, Vol. 15, no.1, 1974.
2. John G. Collier and Geoffrey F. Hewitt, "Introduction to Nuclear power," Hemisphere publishing Corporation, Springer-Verlag, 1987.
3. James H. Rust, "Nuclear power Plant Engineering," Haralson Publishing Company, 1979.

4. Arthur R. Foster, and Robert L. Wright, Jr., “Basic Nuclear Engineering,” Allyn and Bacon, Inc., 1978.