Alternatively, pellets of boron-silicate glass encapsulated in hollow stainless-steel tubes may be used.

2.10 HTS Design Requirements and Engineering Considerations

[GAR96]

2.11 Power Reactor Types

[GAR96]

Table 2.11-1 shows typical characteristics of the fuel for six reference power reactor types [Source TODKAZ90]. The table provides general information about fuel particles, pins, assembly, moderator, and manufacturer.

Table 2.11-1: Typical characteristics of the fuel for six reference power reactor types

Table 1-3 Typical characteristics of the fuel for six reference power reactor types

Characteristic	BWR	PWR(W)	PHWR	HTGR	AGR	LMFBR[a]
Reference design						
Manufacturer	General Electric	Westinghouse	Atomic Energy of Canada, Ltd.	General Atomic	National Nuclear Corp.	Novatome
System (reactor station)	BWR/6	(Sequoyah)	CANDU-600	(Fulton)	HEYSHAM 2	(Superphenix)
Moderator	H_2O	H_2O	D_2O	Graphite	Graphite	—
Neutron energy	Thermal	Thermal	Thermal	Thermal	Thermal	Fast
Fuel production	Converter	Converter	Converter	Converter	Converter	Breeder
Fuel[b]						
Particles						
Geometry	Cylindrical pellet	Cylindrical pellet	Cylindrical pellet	Coated microspheres	Cylindrical pellet	Cylindrical pellet
Dimensions (mm)	10.4*D* × 10.4*H*	8.2*D* × 13.5*H*	12.2*D* × 16.4*H*	400–800 μm *D*	14.51*D* × 14.51*H*	7.0 *D*
Chemical form	UO_2	UO_2	UO_2	UC/ThO_2	UO_2	PuO_2/UO_2
Fissile (wt% 1st core ave.)	1.7 ^{235}U	2.6 ^{235}U	0.711 ^{235}U	93 ^{235}U	2.2 ^{235}U	15–18 ^{239}Pu
Fertile	^{238}U	^{238}U	^{238}U	Th	^{238}U	Depleted U
Pins						
Geometry	Pellet stack in clad tube	Pellet stack in clad tube	Pellet stack in clad tube	Cylindrical fuel stack	Pellet stack in clad tube	Pellet stack in clad tube
Dimensions (mm)	12.27*D* × 4.1 m*H*	9.5*D* × 4 m*H*	13.1*D* × 490*L*	15.7*D* × 62*L*	14.89*D* × 987*H*	8.65*D* × 2.7 m*H*(C) 15.8*D* × 1.95 m*H*(BR)
Clad material	Zircaloy-2	Zircaloy-4	Zircaloy-4	Graphite	Stainless steel	Stainless steel
Clad thickness (mm)	0.813	0.57	0.42	—	0.38	0.7
Assembly						
Geometry[c]	8 × 8 square rod array	17 × 17 square rod array	Concentric circles	Hexagonal graphite block	Concentric circles	Hexagonal rod array
Rod pitch (mm)	16.2	12.6	14.6	—	25.7	9.7 (C)/17.0 (BR)
No. rod locations	64	289	37	132 (SA)/76 (CA)[d]	37	271 (C)/91 (BR)
No. fuel rods	62	264	37	132 (SA)/76 (CA)[d]	36	271 (C)/91 (BR)
Outer dimensions (mm)	139	214	102*D* × 495*L*	360*F* × 793*H*	190.4 (inner)	173*F*
Channel	Yes	No	No	No	Yes	Yes
Total weight (kg)	273	—	—	—	342	—

Source: Knief [4] except AGR-HEYSHAM 2 data are from Alderson [1], and LMFBR pin and pellet diameters are from Vendryes [5].
[a]LMFBR-core (C), radial blanket (BR), axial blanket (BA).
[b]Fuel dimensions: diameter (*D*), height (*H*), length (*L*), (across the) flats (*F*), (width of) square (*S*).
[c]LWRs have utilized a range of number of rods.
[d]HTGR-standard assembly (SA), control assembly (CA).

2.11.1 "Magnox" Reactors

[GAR96]

2.11.2 AGR

[GAR96]

2.11.3 HTGR

[GAR96]

2.11.4 PWR

[GAR96]

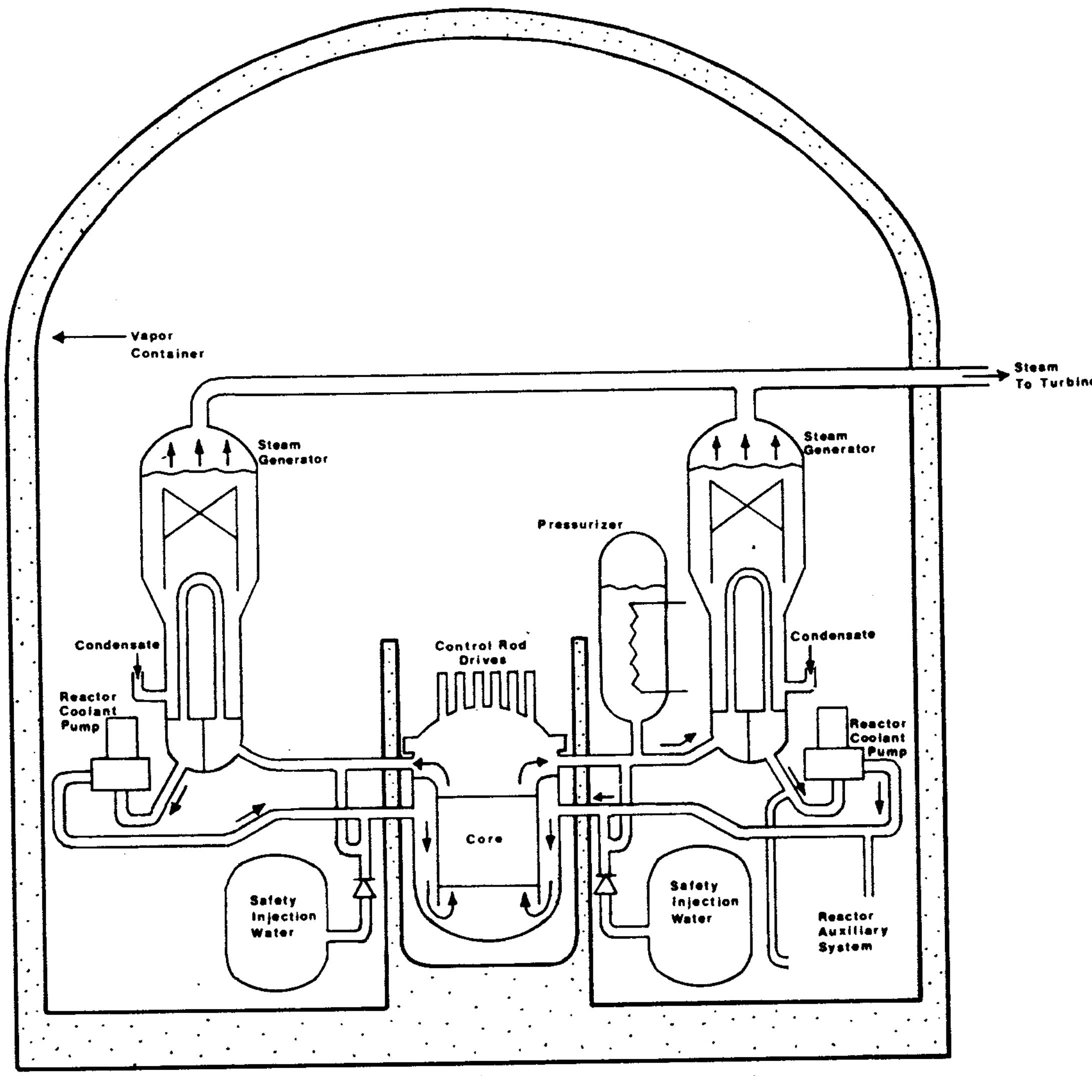


Fig. 1.1(a). Pressurized water reactor system — vessel concept.

Figure 2.11-1: PWR reactor system [WIES77]

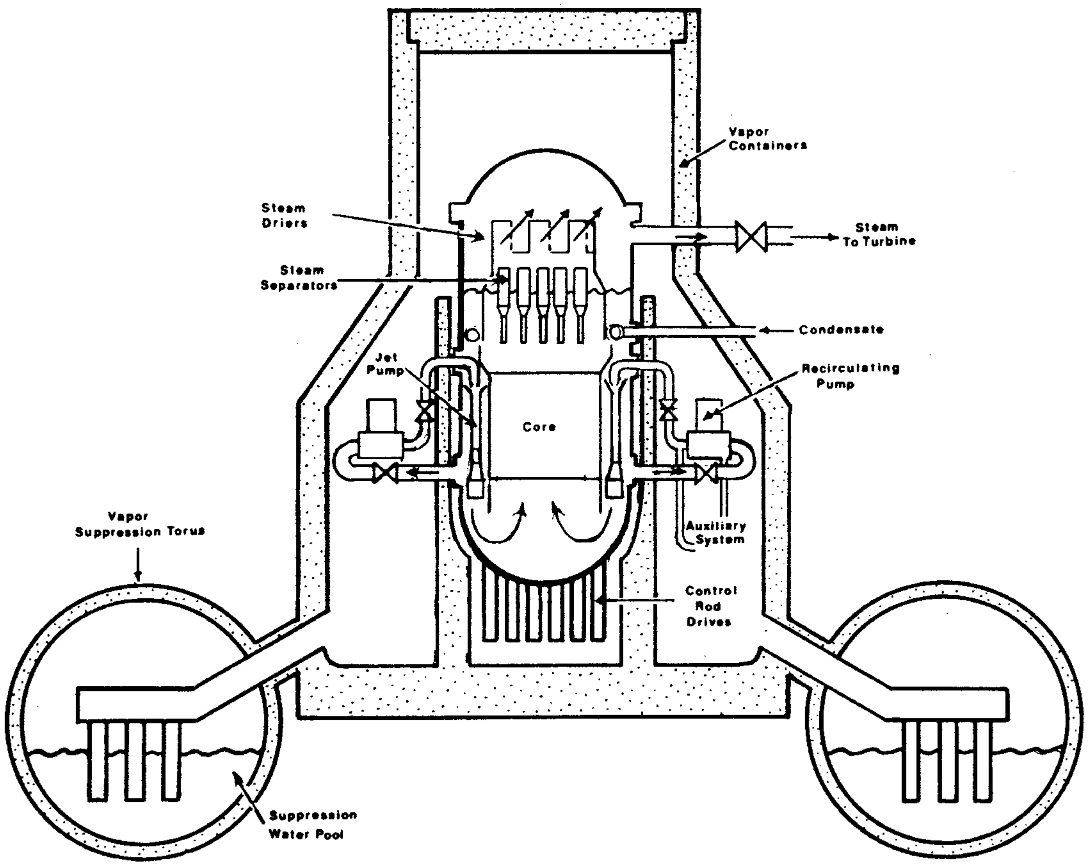


Figure 2.11-2: BWR reactor system

2.11.5 BWR

The water enters the BWR core at near saturation temperature so that nearly all the heat added in the reactor is used for generating steam. The exiting steam-water mixture flows through a set of steam separators located above the core. The wet steam from the separators is dried and then proceeds directly to the turbine. The unevaporated water discharged by the steam separators is recirculated to the core inlet through the annular region around the core by means of a set of jet pumps. Water is pumped from the lower end of the annulus by an external pump and returned to the reactor vessel as a high-velocity stream at the jet pump throat. In this system, the external pumps circulate only a fraction of the water circulated through the core.

The external recirculating pumps are provided with variable-speed drives and thus the coolant flow may be increased when more power is required. By varying the flow, the steam volume in the core can be decreased slightly as power increases. This leads to a reactivity increase which compensates for the decrease in reactivity due to increased fuel temperature with increased power (Doppler Effect). Using this mechanism, power variation between ~70% and 100% of full load can be handled without control rod movement. The control rods are inserted from the bottom of the core. The bottom insertion of the control helps flatten the power profile axially. In the absence of the control rods, power would be higher in the lower portion of the core because of the lower steam content.

A containment structure is provided, but it is usually much smaller compared to the PWR's. The reactor is surrounded by a light-bulb-shaped container which is connected to a torus containing a water pool. Large pipes, placed circumferentially around the container, would conduct any steam released to the vapor suppression pool.

2.11.6 LMFBR

[GAR96]

2.11.7 CANDU

[GAR96]

2.11.8 CANDU-PHW

[GAR96]

Figure 2.11-4 shows a typical CANDU-6 plant design. Figure 2.10 in Reference [GAR96] shows the CANDU design used in the Pickering A station with the typical moderator dump system. This system was not included in any of the later CANDU designs.

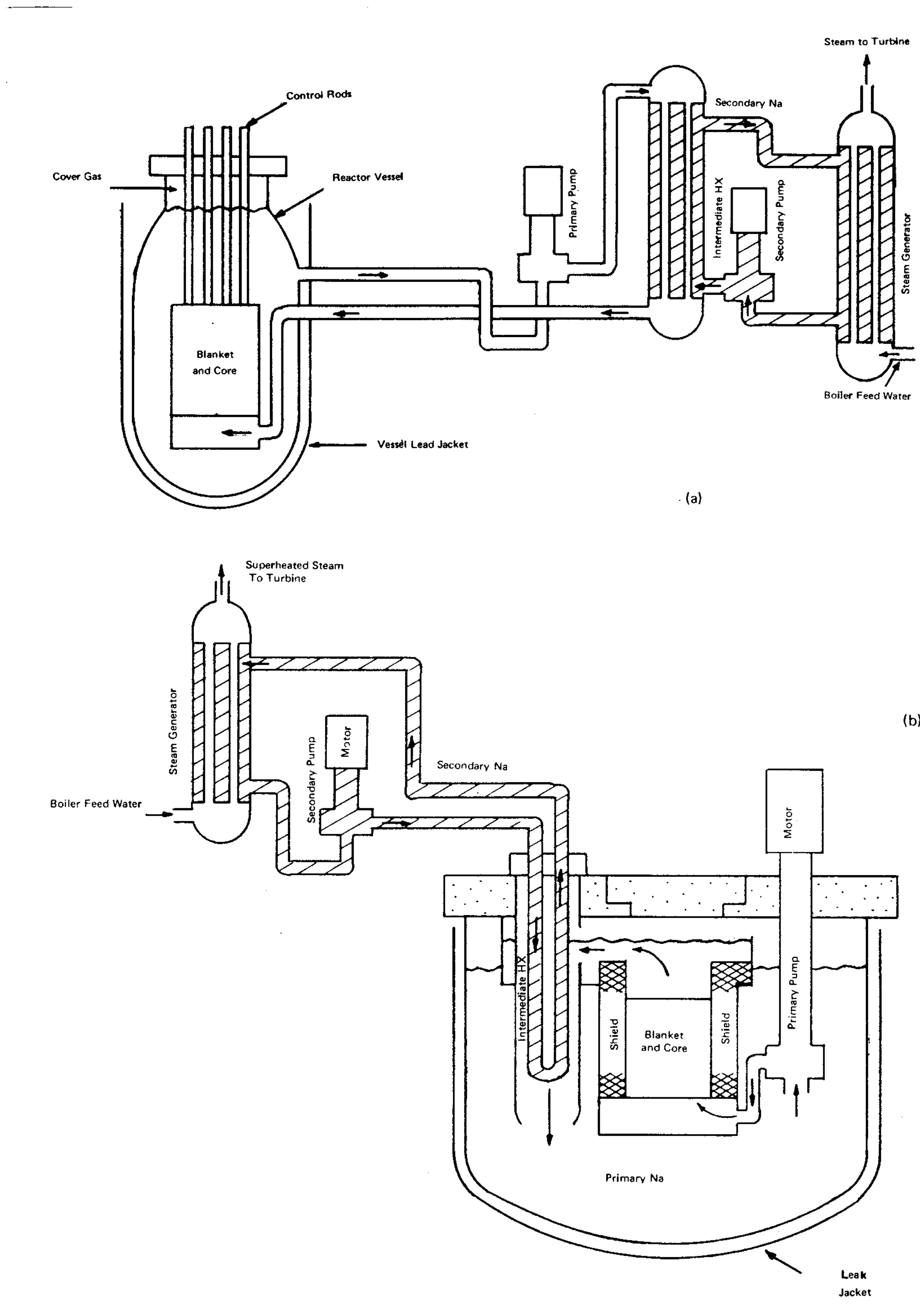


Figure 2.11-3: LMFBR reactor system

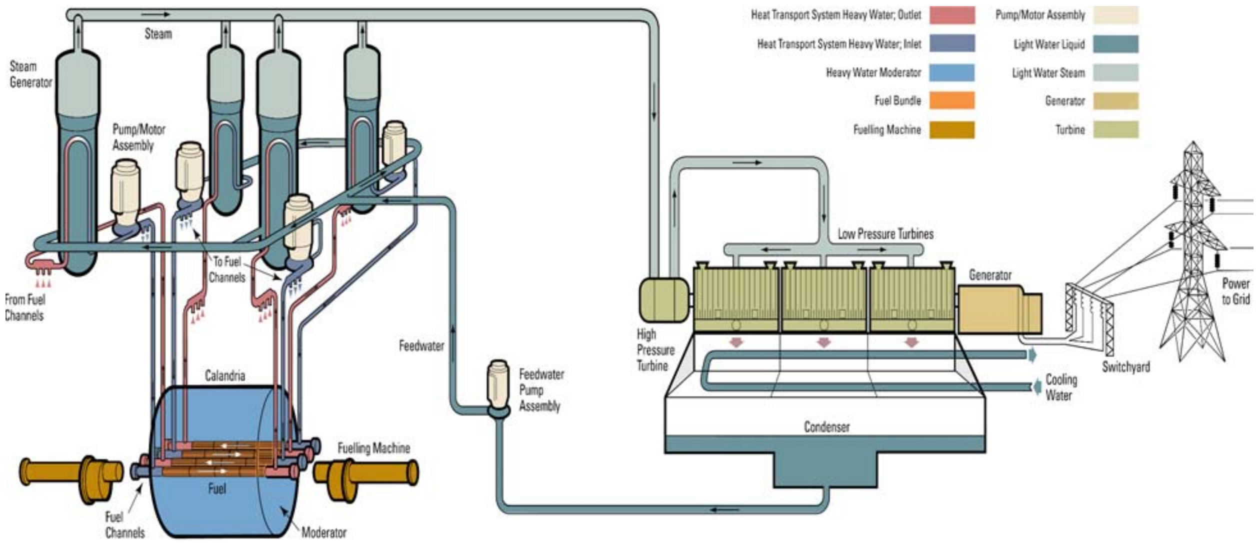


Figure 2.11-4: CANDU-PHW reactor system

2.11.9 CANDU-BLW [AECL97]

Figure 2.11-5 shows the CANDU-BLW (Boiling-Light-Water) reactor system, whereas Figure 2.11-6 shows a cross section of the reactor building for the CANDU-BLW. This reactor was designed to use natural uranium.

In the CANDU-BLW design the pressure tubes were oriented vertically in the core and the fuel bundles, which were strung on a central support tube, could be inserted and removed from the bottom only. As the light water coolant passed upward through the core it boiled and exited the core at about 290°C, 7.7 MPa and 20% quality. The coolant was then directed to a steam separator from which the steam passed directly to the turbine generator. The water from the separator was mixed with the steam condensate from the turbine generator and then recirculated.

The potential advantages of the CANDU-BLW reactor design is the lower capital cost because of less usage of heavy water., no steam generators, and a slightly higher thermal efficiency. Also, it would have lower operating cost because of eliminating the problem of heavy water leakage. One of the difficulties with this design was worse neutron economy (light water), resulting in lower fuel burnup in natural uranium fuel.

The main difficulty with this design was reactor control, resulting from a highly positive power coefficient. Even at a constant reactor power, it was impossible to always keep at constant steam quality in each channel; when the local quality increased, the local neutron flux increased and the local fuel power increased, driving the local steam quality even higher. These local reactivity excursions would not necessarily lead to a loss of overall reactor control, as the changes in spatial flux shape would increase overall neutron leakage. Nevertheless, the resulting spatial power oscillations were expected to be intolerable. Thus it was deemed essential to have a spatial flux system to limit them to acceptable levels.

In 1965 AECL and Hydro-Quebec started a project Gentilly-1 that was supposed to use a prototype CANDU-BLW design with 250 MW power output. The first power from this plant was produced in April 1971, and full power reached in May 1972. But the station did not perform as designed and expected, resulting in station shutdown. However, other stations around the world with a similar design operated successfully for many years, such as the British Steam-Generating Heavy-Water Reactor (operating until 1990), and the Japanese equivalent FUGEN, which is still operating.

The engineers and designers of the CANDU-BLW design have recognized that the enriched ^{235}U or plutonium-enriched fuel, rather then natural uranium, would have been a better choice. With enriched fuel, the neutron economy would have been better, allowing thinner fuel pins, and hence higher channel power, tighter lattice pitch, resulting in considerably lower inventory of heavy water. With enrichment, the fuel burnup could have been up to 580 MWh/kgU, rather then only 180 MWh/kgU. Most importantly, the enriched fuel, the positive power coefficient would have been significantly reduced, making it much easier for the control rods to correct any reactivity excursions resulting from boiling imbalance.

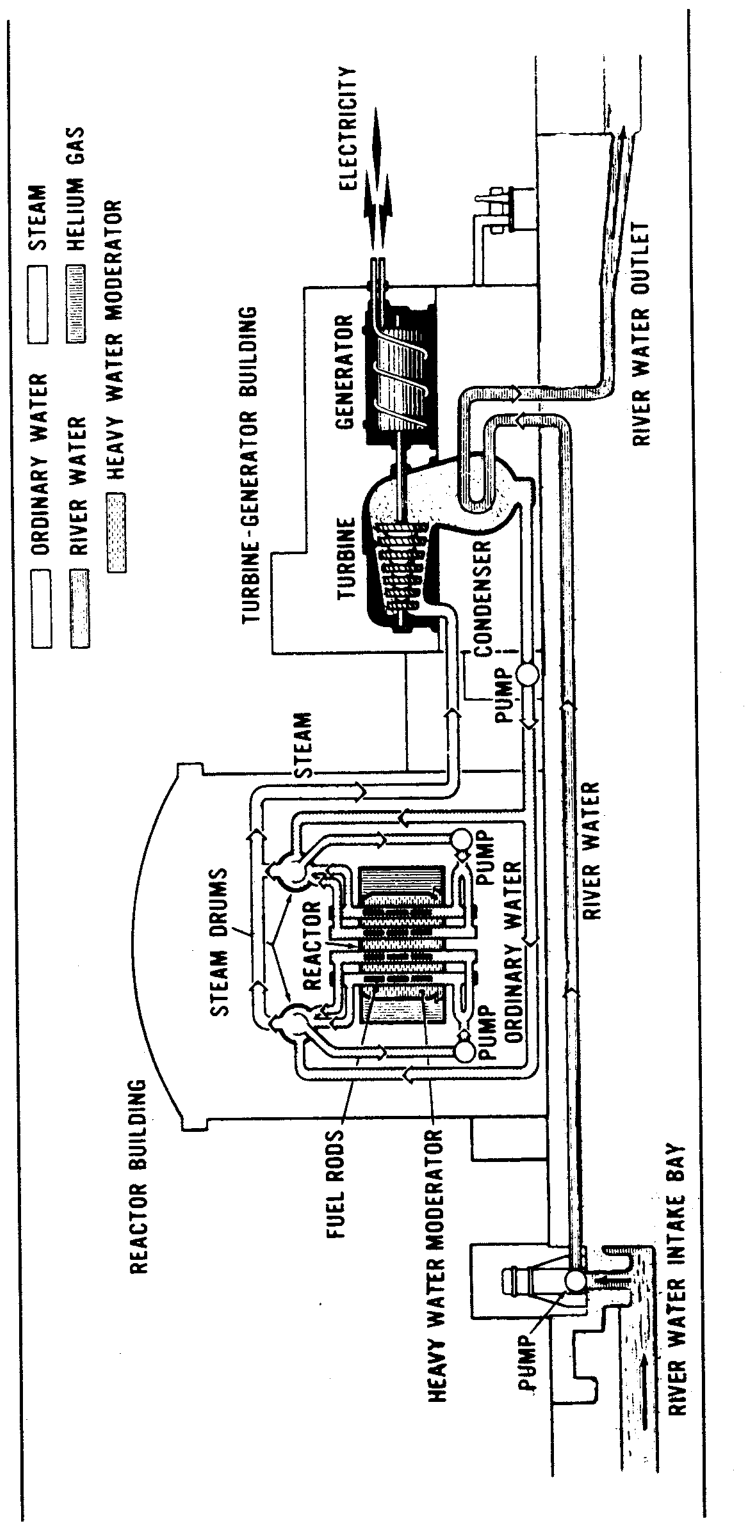


Figure 2.11-6: Schematic illustration of a CANDU-BLW power station

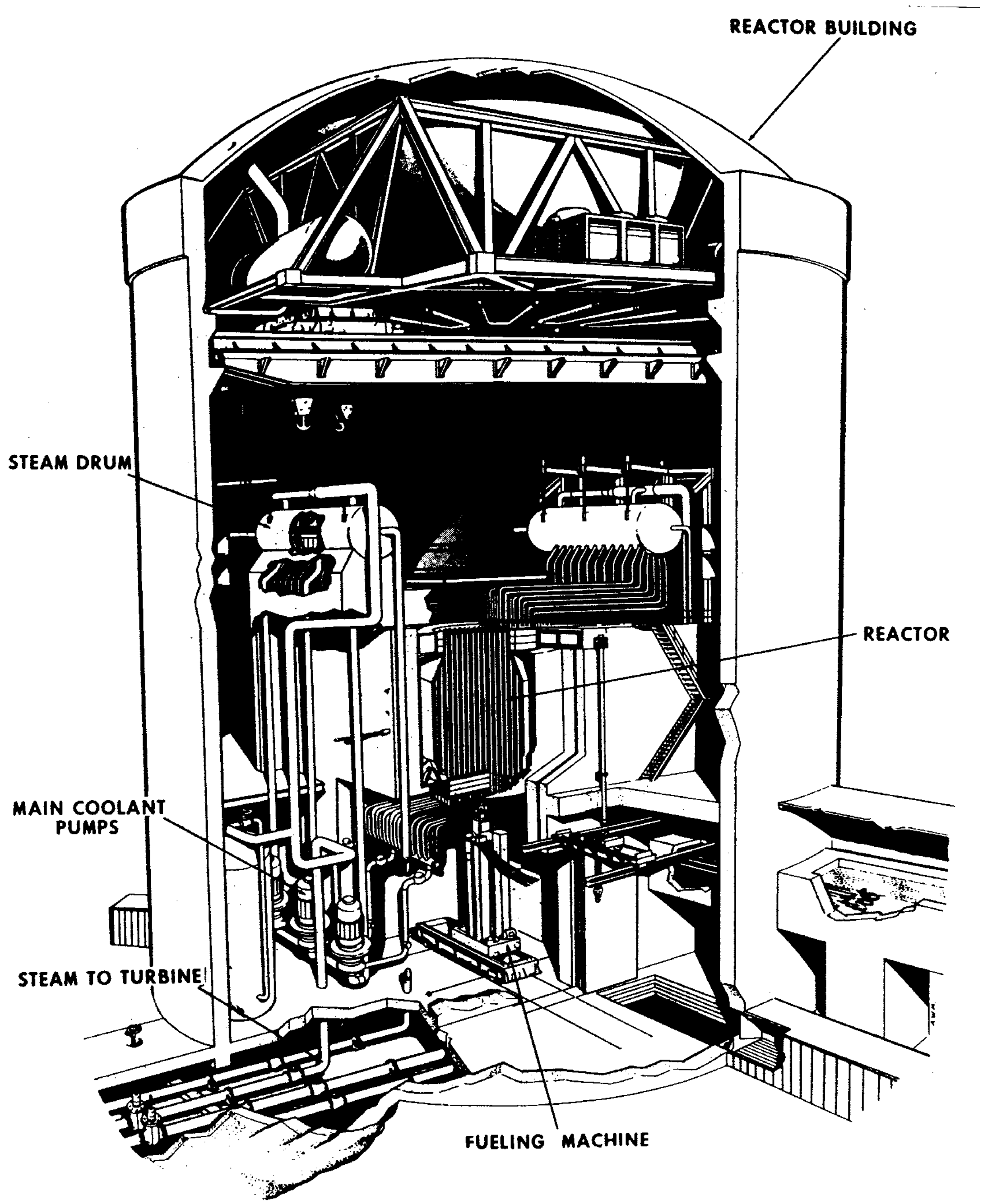


Figure 2.11-6: Cutaway of a CANDU-BLW reactor building

2.11.10 CANDU-OCR [AECL97]

Figure 2.11-7 shows the CANDU-OCR (Organic-Liquid-Coolant) reactor system, whereas Figure 2.11-8 shows a cross section of the reactor building for the CANDU-OCR. This reactor was designed to use natural uranium. A research reactor WR-1 was built at Whiteshell Laboratories to support the research and design.

The conceptual natural-uranium-fueled CANDU-OCR was similar to the CANDU-BLW design, in that the pressure tubes were oriented vertically in the calandria and the fuel bundles were strung on a central support tube. The fueling was from the top rather than from the bottom. The steam generating system was very similar to the current CANDU-PHW design. The potential advantages of the CANDU-OCR were lower capital cost, resulting from lower heavy water inventory, a lower coolant pressure and a higher coolant temperature (higher thermal efficiency), and its lower operating cost resulting from the elimination of the heavy-water leakage problem. These advantages were offset by the higher fueling costs, that were believed to be resolved by usage of carbide fuel.

The main difficulty with the CANDU-OCR design was gradual decomposition of the coolant under heat and radiation. This was expected to be capable of foul fuel surfaces and reduce heat transfer rate. Therefore, a lot of research was devoted to identify organic coolants that had lower rate of decomposition, and thus lower rate of fouling. This research was carried out at WR-1 reactor.

The final design of the CANDU-OCR that was authorized by the AECL board was using Zr-2.5Nb-clad natural uranium carbide fuel, with pressure tubes from Zr-2.5Nb and coolant was HB-40. The coolant outlet temperature was 400°C, which led to plant thermal efficiency of about 34%. The station had some attractive safety features compared to the water-cooled reactors, including low radiation fields, low stored energy, low coolant pressure, no fuel ballooning and no dryout. However, the coolant was flammable, thus leading to increased fire hazard. Based on this design study, the CANDU-OCR would have a 10% advantage over the CANDU-PHW on both capital and unit energy costs.

AECL decided to cancel the CANDU-OCR project in 1973, based on the good marketing potential from the CANDU-PHW design.

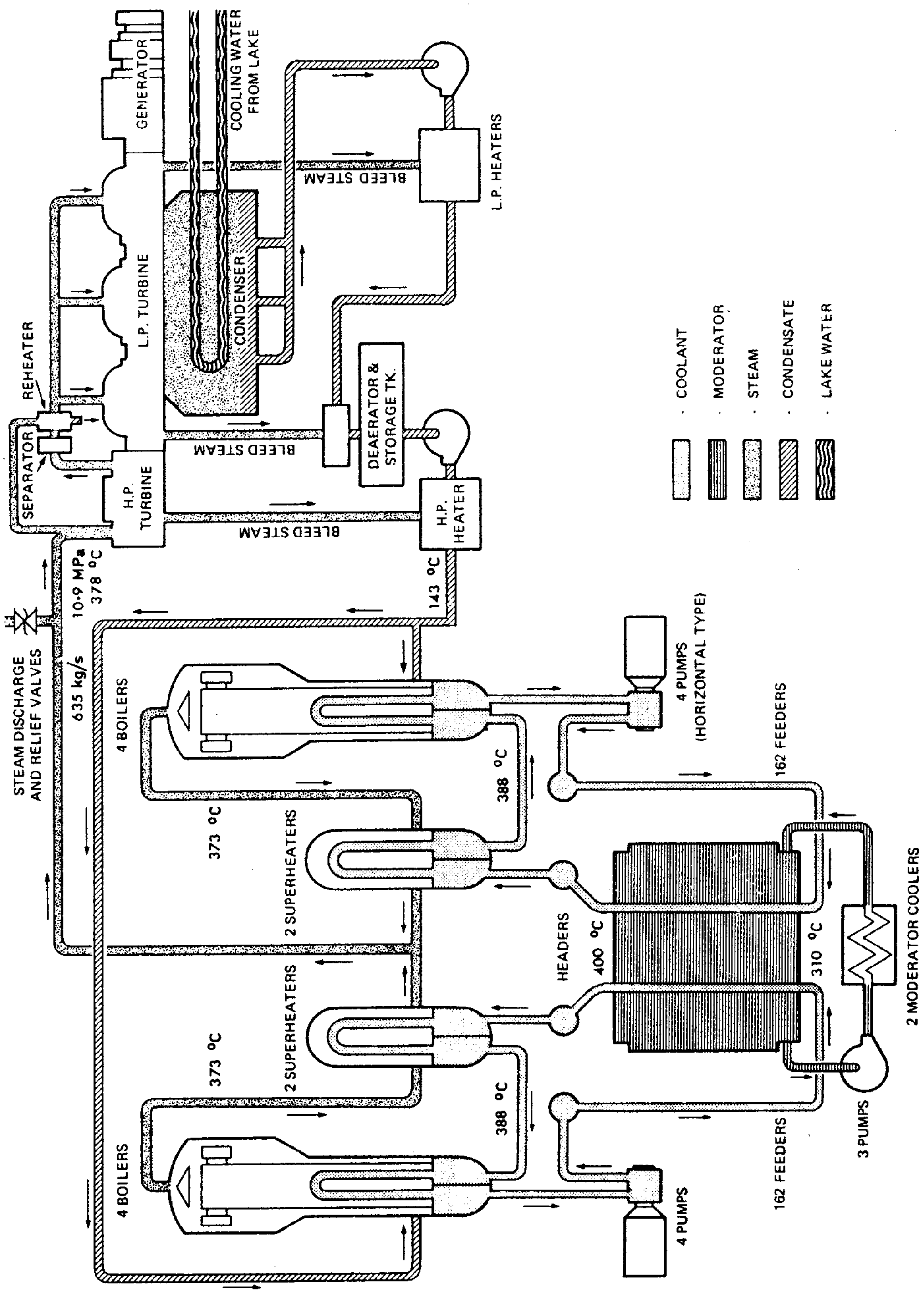
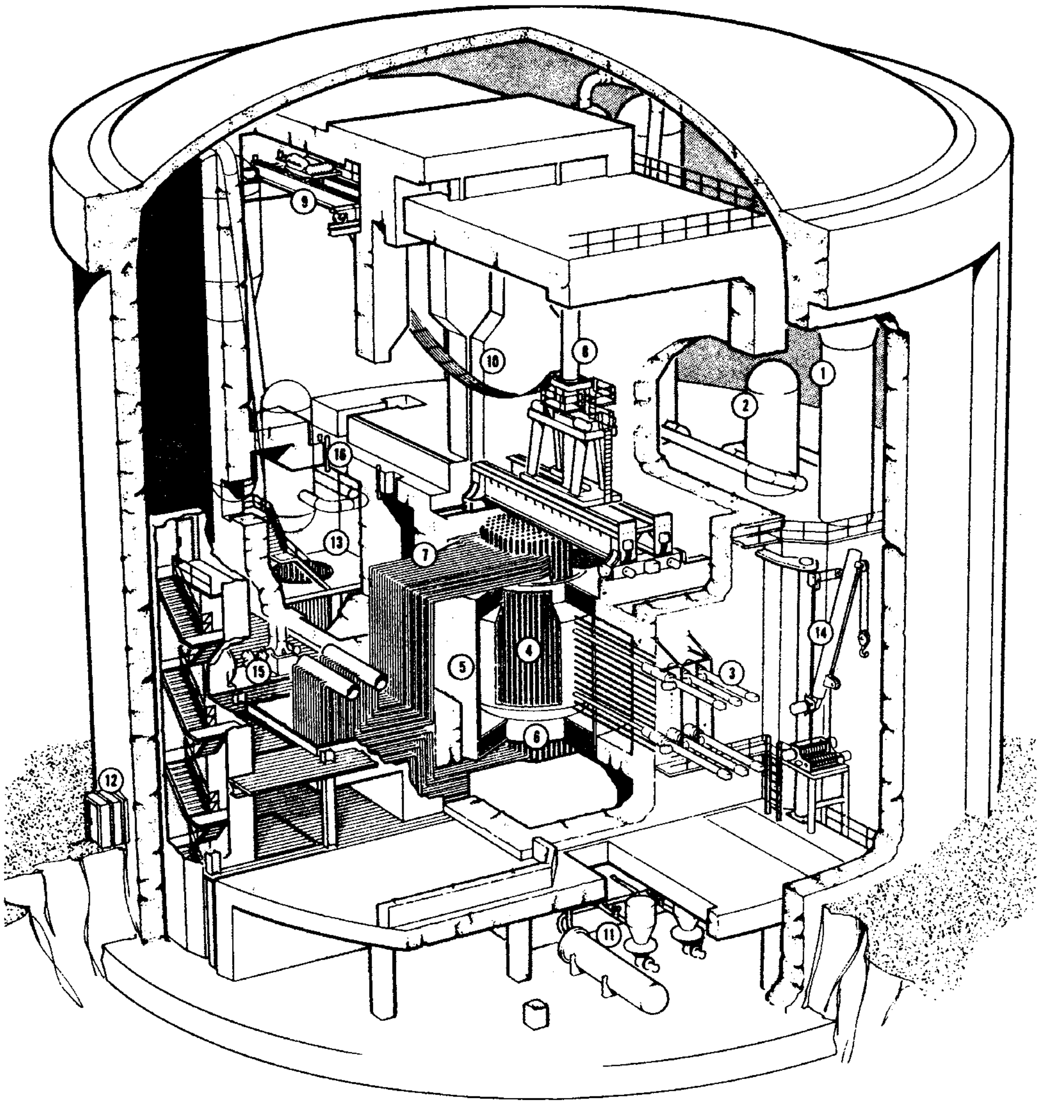


Figure 2.11-7: Schematic diagram of a CANDU-OCR power station



1. Boilers (8)
2. Superheaters (4)
3. Booster Rods
4. Calandria Assembly
5. Shield Tank
6. End Shield
7. Feeders
8. Fueling Machine
9. F/M Service Crane
10. F/M Vault Door
11. Moderator System
12. Emergency Airlock
13. Fuel Transfer Bay
14. Booster Flask Crane
15. Primary Pumps (4)
16. Fueling Machine Ports

Figure 2.11-8: Cutaway of a CANDU-OCR reactor building

2.11.11 CANDU Future Designs

While continuous evolutionary improvements are being made to the CANDU-PHW through the CANDU-6 and CANDU-9 designs that are currently being offered at the market, work is in progress to develop more revolutionary designs. The three main objectives are being considered in making these design changes:

a) to demonstrate innovation of the design through product development
b) to provide assurance of environmental acceptability that meets increasingly demanding regulations, and
c) to establish competitive economics through reduced capital and operating costs.

To satisfy above objectives, currently the following design changes are being considered:

- eliminate complexity and components (reduce heavy water inventory, etc.),
- reduce the number of components (valves, etc.)
- simplify and upgrade systems to make them more efficient (fueling machines, chemistry systems, etc.),
- increase overall thermal efficiency of the station (increase pressure and temperature of the HTS), and
- demonstrate safety enhancements (reduce positive void reactivity, passive heat transfer improvements, etc.)

The most important characteristics being considered for future CANDU designs are:

- using light water in the HTS.
- slight fuel enrichment in ^{235}U.
- reduction of the channel lattice pitch
- the positive void reactivity coefficient is reduced.
- the calandria vessel size is significantly reduced:
  - ⇒ CANDU 6 with 380 channels has inner diameter of 760 cm
  - ⇒ CANDU 3 with 232 channels has inner diameters of 636 cm
  - ⇒ CANDU future designs will tend to reduce inner diameter below 500 cm
- the channel maximum power is increased to about 8 MW.
- turbine outlet temperature increased, thus providing thermal efficiency.
- simplified HTS and ECC design (more reliable ECC injection with simplified interface; no risk of mixing light and heavy water).
- reduced risk of tritium leakage (usage of light water).
- simplified single-ended refueling machine.
- passive reactor regulating system by usage of moderator as reactor control system (allowing moderator boiling to reduce moderation efficiency).
- passive reactor shutdown system by using moderator.
- enhanced thermosyphoning capability of the HTS and moderator.

2.11.12 A Comparison Between CANDU Reactors and Other Types

[GAR96]