

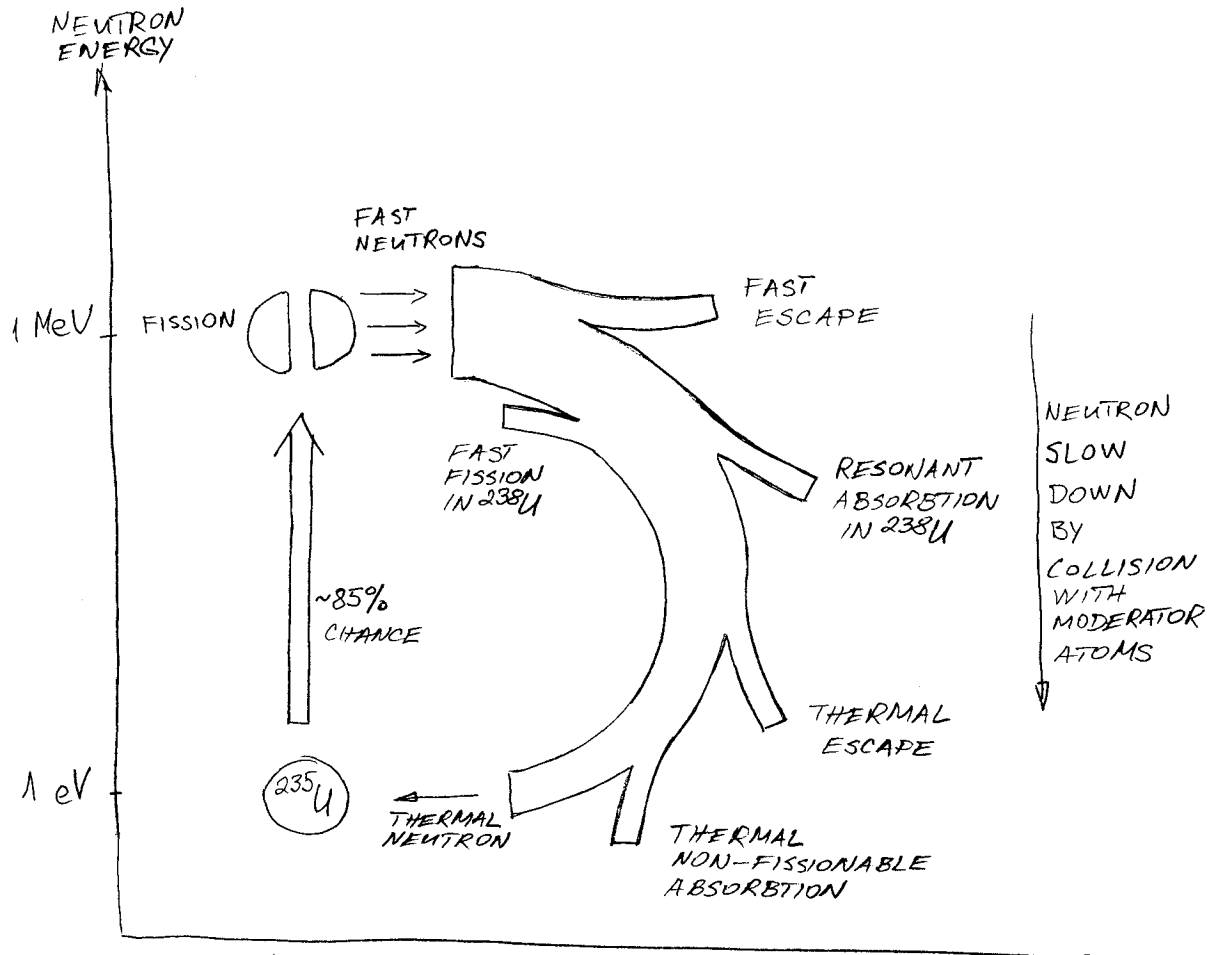
2. DESIGN ELEMENTS AND ENGINEERING CONSIDERATIONS

2.1 Introduction

[GAR96]

2.2 Basic Neutron Cycle

[GAR96]



$$1 \text{ (eV)} = 1.602 \cdot 10^{-19} \text{ (J)} = 4.4 \cdot 10^{-26} \text{ (kWh)}$$

Figure 2.2-1 The basic neutron cycle

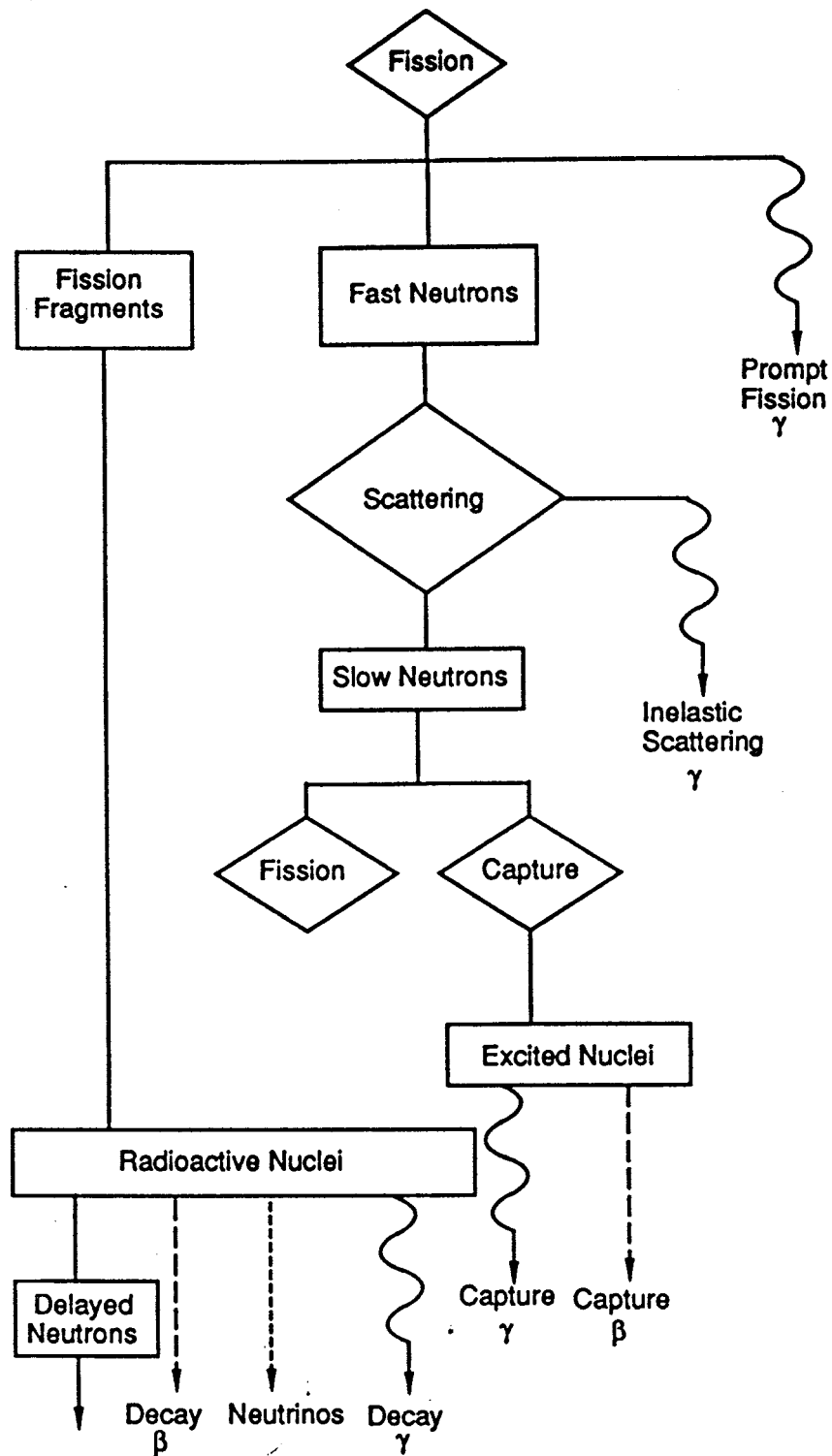


Figure 2.2-2 Forms of energy release in a reactor [TODKAZ90]

### 2.3 Possible Fuels

[GAR96]

### 2.4 Heat Transfer Considerations

[GAR96]

### 2.5 Uranium Fuel Forms

[GAR96]

#### 2.5.1 Metallic Fuels

Unalloyed uranium metal is a very poor reactor fuel as it exhibits substantial growth under irradiation. Highly irradiated specimens have been known to show axial growth equaling more than 60% of the original sample length. Most of the uranium's irradiation instability can be attributed to the properties of the  $\alpha$  phase, which is the phase present below 663°C.

Uranium metal is highly reactive chemically and can be used at high temperatures with only a few coolants, e.g. carbon dioxide and helium.

Low-alloy uranium fuel (with small amount of alloying material) has been demonstrated to increase corrosion resistance to high-temperature water and to improve irradiation stability. Good results are obtained with  $U_3Si$  (U-3.8 wt.% Si). Irradiation stability is improved by still less than desired. Corrosion resistance is about 500 times greater than that of unalloyed uranium. More satisfactory fuels can be obtained by addition of alloying materials which allow the  $\gamma$  phase to be retained at operating condition. This can be done by addition of 10-20 wt.% Mo or Nb. Alloys containing 10-13.5% molybdenum appeared most promising.

Fuels which are highly satisfactory from both the irradiation-damage and water-corrosion standpoints can be obtained with large additions of alloying material. The only high-alloy uranium fuels having power-reactor application are the alloys of uranium and zirconium because of the good neutron economy features of the zirconium.

#### 2.5.2 Ceramic Fuels

Uranium dioxide is the most widely used of all reactor materials. It has an irradiation stability far superior to that of metallic fuels, and it is capable of withstanding the high burnups required for economic power-reactor application. When properly prepared, it exhibits excellent resistance to high-temperature water or sodium.

Uranium dioxide is obtained from the gaseous  $UF_6$  product of the diffusion plants. Hydrolysis of  $UF_6$  produces  $UO_2F_2$ , which is reacted in a dilute ammonia solution to form a precipitate of ammonium diuranate. This precipitate is calcined to  $UO_3$ , which is then reduced to  $UO_2$  by hydrogen at about 800°C. The oxide is produced as fine powder, which needs to be pressed into a high-density compact followed by sintering at high temperature into cylindrical pellets.

The thermal conductivity of the uranium dioxide is quite low. At the high power output required in the power-reactor service, this leads to very high temperature gradients across the fuel element. The thermal stresses which are caused generally lead to radial cracking of the pellets during operation. However, the cracking does not appear to cause any deterioration in fuel

performance providing the pellets are suitably restrained by a cladding. However, the fuel element centerline melting remains to be of concern, and it imposes an important safety margin criterion.

The chief limitation on the performance of  $\text{UO}_2$  fuel is the swelling caused by gaseous fission products. At low and moderate burnups, the swelling is slight and roughly linear with burnup. Above a critical burnup, the swelling increases markedly and continued exposure of the fuel leads to unacceptable dimensional changes. The critical burnup is primarily a function of fuel density, a value of about  $17 \cdot 10^3$  MWd/ton being obtained with fuel of 97% the oxide density, and a value of the order of  $42 \cdot 10^3$  MWd/ton being obtained with 93% dense fuel.

Thorium dioxide ( $\text{ThO}_2$ ) behaves similarly as  $\text{UO}_2$  under irradiation. Thermal stress induced cracking is also observed in the  $\text{ThO}_2$  fuels.

Carbide fuels are primarily intended for use in fast-breeder reactors (provide better breeding ratios). Since these reactors operate on the U-Pu cycle, the fuel consists of mixed uranium and plutonium oxides. Thermal conductivity of the UC is substantially better than that of  $\text{UO}_2$ . Hence, considerably higher specific powers are possible without approaching the centreline melting. However, at higher temperatures the swelling rate becomes excessive. The higher thermal conductivity of the carbide fuel leads to a much lower Doppler coefficient than obtained with an oxide core. Since Doppler coefficient is the main component of negative reactivity feedback in most accident situations, the design of a safe carbide-fueled core may be more difficult than that of an oxide core.

Nitride fuels have also been considered for use in fast reactor fuel.

### 2.5.3 Dispersion Fuels

In a dispersion-type fuel, particles of fissile material are imbedded in a metallic ceramic matrix. Such fuels can generally withstand significantly higher burnups than alloy fuels. If the fuel particles are separated sufficiently, the areas damaged by fission fragments will not overlap and there remains a continuous metal phase.

$\text{UO}_2$  and  $\text{PuO}_2$  can be dispersed in aluminum, stainless steel and zirconium alloys. The low-temperature limitation on the use of aluminum eliminates these dispersions from consideration for use in power reactors. While stainless steel suffers from no such limitation, poor neutron economy eliminates it from consideration in commercial reactors. Dispersion of  $\text{UO}_2$  in zirconium alloys form highly satisfactory fuels if highly enriched uranium is to be used. Fuel performance is still limited by fuel swelling.

The most important dispersion fuel used is the dispersion of mixed uranium-thorium carbides or oxides in graphite. These dispersion are used in the high-temperature gas-cooled reactors. The dispersion is achieved by multiple layers of pyrolytic carbon.

## 2.6 Fuel Claddings

### 2.6.1 Aluminum

Aluminum has been used for cladding of fuel elements cooled by low-temperature water. When the water temperature is raised to the levels needed for power reactor production, aluminum corrosion rates become excessive. Relatively low melting temperatures cause a concern when aluminum cladding fuel elements are exposed to accident conditions.

### 2.6.2 Stainless Steel

The 300-series stainless steels (approximately 18% Cr, 8% Ni) have both high-temperature strength and excellent corrosion resistance. Stainless steels have relatively good resistance to irradiation. Although they lose ductility under irradiation, the decrease in ductility is less than for carbon steel. An important problem with stainless steels is the density decrease (swelling) at high irradiation (up to 10% swelling was observed in flux of  $10^{23}$  n/cm<sup>2</sup>, typical for fast reactors).

Stainless steel cladding has a significant impact on the neutron economy due to the relatively high cross-section of steel for thermal neutrons. This is the main reason for stainless steel being avoided as cladding material in thermal power reactors.

Stainless steels as cladding is used in reactors where high-temperature service is needed. This type of cladding was used in the British high-temperature, CO<sub>2</sub>-cooled reactors. Both gas-cooled and sodium-cooled fast-reactor designs use stainless steel cladding.

### 2.6.3 Stainless Steel

The primary advantage of zirconium as cladding material is the very low cross-section for thermal neutrons, which greatly improves neutron economy. Zirconium has a very good water corrosion resistance at high-temperatures. Adding tin, iron and chromium to zirconium greatly improves the mechanical features of zirconium. The most well known zirconium alloys are Zircaloy 2, 3 and 4, which found wide application as cladding for the power reactor fuel elements.

Zirconium alloys are unsuitable at very high temperatures even though the melting point of zirconium is 1852°C. At 862°C zirconium goes from a close-packed-hexagonal structure to one that is body-centered-cubic, and it is necessary to stay below this phase change. At these temperatures of the zirconium cladding interface a reaction with the UO<sub>2</sub> can occur.

Zirconium alloys exhibit significant creep at the temperature and stresses typical for PWR reactor design. Creep rates increase markedly with temperature and are accelerated by reactor irradiation.

At high temperatures (above 800°C), zirconium reacts with steam to release hydrogen in an exothermic reaction. This reaction must be considered in evaluation of any LOCA event which could lead to exposing the fuel elements to steam. The reaction of an appreciable fraction of the clad could add significantly to the severity of the accident. Also, the mechanical properties of the ZrO<sub>2</sub> are not as favorable compared to the zirconium.

The last point to be made about the zirconium cladding is the cost. Zirconium tubes are more expensive than the stainless-steel tubes. Fabrication costs are also higher since all welding must be done in an inert atmosphere. However, the decrease in fuel costs more than offsets the increased material and fabrication costs.

## 2.7 Reactor Coolants

[GAR96]

### 2.7.1 Ordinary Water and Heavy Water

The properties of ordinary water make it both excellent moderator and coolant. The small neutron migration length allows the light-water-moderated cores to be very compact. The relatively.

Ionizing irradiation (including  $\gamma$ 's released as a result of neutron-radiative capture) causes decomposition (radiolysis) of water. Decomposition occurs in both H<sub>2</sub>O and D<sub>2</sub>O via the same mechanism. The rate of gas evolution (radiolysis) is proportional to the radiation flux and decreases with increasing temperature.

Radiolysis of water is an important phenomenon that needs to be considered in analysis of LOCA events in power reactor. Hydrogen is produced with the radiolysis of water discharged from the break, and can accumulate in the reactor building. The radiolysis of the discharged water can occur as a result of the  $\gamma$  radiation from the sources within the discharge water (high contribution), or from the reactor fuel (low contribution). Above a certain concentration, hydrogen will burn at an explosive rate impose high risk to reactor components and building structures. The accumulation of hydrogen produced by radiolysis of discharged water in a LOCA event can be controlled by passive autocatalytic recombiners.

Water circulated through a reactor core will exhibit appreciable induced radioactivity. The  $\gamma$  activity induced is primarily due to the 7.4 s half-life <sup>16</sup>N produced by fast-neutron interaction with <sup>16</sup>O. Additional activity may be introduced by activation of dissolved impurities and dissolved or suspended corrosion products. Hence, water purification is an important process that is continuously performed by dedicated reactor systems.

### 2.7.2 Gaseous Coolants

A good gaseous coolant should have a low neutron-absorption cross-section, high heat capacity and high thermal conductivity. Hydrogen satisfies these characteristics, but it is hazardous gas with respect to explosive burn should it come into contact with oxygen. Methane is also a good candidate, but it is unstable under irradiation.

Carbon dioxide is a good coolant because it has low cross-section, is inert at low and moderate temperatures, and is non-toxic. It is also inexpensive, and leakage is not a major cost concern. The maximum temperatures which can be attained with CO<sub>2</sub> are severely limited. The reduction of CO<sub>2</sub> in CO by graphite at high temperatures, and the oxidation of carbon steel by CO<sub>2</sub> are well known.

Helium would appear as a good gaseous coolant because it has high thermal conductivity and specific heat. It has extremely low cross-section for neutron absorption, and it is chemically inert and non-hazardous. However, it is expensive and in relatively limited supply. Any significant leakage leads to an appreciable operating cost.

All gaseous coolant have a disadvantage that a significant fraction of the plant energy output must be used for circulating the coolant. Furthermore, the low heat transfer coefficient requires a large heat-transfer area. This may require a larger core or the use of extended surfaces (fins).

### 2.7.3 Liquid Metal Coolants

The importance of the liquid metal coolants is only for the fast reactors due to the characteristic of the metals to be poor neutron moderators. Sodium is one of the most suitable liquid-metal coolant.

Sodium becomes radioactive due to the formation of  $^{24}\text{Na}$  by neutron capture. This radioisotope has a 15-h half-life and emits gamma rays of 1.37 and 2.75 MeV. Hence, shielding of the cooling system is necessary. Care must be taken to ensure a leak-tight system. Sodium is chemically reactive and on exposure to air will burn with evolution of the oxide as dense smoke. Further, it violently reacts with water, producing NaOH and hydrogen gas.

## 2.8 Neutron Moderators

[GAR96]

### 2.8.1 Moderating Arrangements

[GAR96]

## 2.9 Control Materials

Reactor control is most commonly accomplished by

- movement of rods in the reactor core, or
- injection of liquid into the coolant and/or moderator,

containing material (poisons) with high neutron-absorption cross-section in the thermal and near-thermal range. This control mechanism is not as effective in the fast reactors as it is in the thermal reactors. Fortunately, the change in reactivity with lifetime in a fast reactor is much lower than in a thermal reactor, and hence substantially less control is needed.

### 2.9.1 Hafnium

Hafnium is probably the best control-rod material for water-cooled reactors. It is found together with zirconium, and comes as by product from the separation process for zirconium. Hafnium is chemically similar to zirconium and shows the same high resistance to corrosion by high-temperature water.

Hafnium consists of four isotopes each of which have appreciable absorption cross-section. Note that the capture of neutrons by one isotope leads to the production of the next higher isotope, which is also an effective absorber. Thus hafnium remains effective poison for a long time.

### 2.9.2 Silver-Indium-Cadmium Alloys

By alloying cadmium, which has a high thermal-absorption cross-section, with silver and indium, which have high resonance absorption, a highly effective absorber is produced. The alloy is composed by 80% Ag, 15% In and 5% Cd, and can be readily fabricated and have adequate strength at water-reactor temperatures.

These alloys exhibit moderate resistance to corrosion by hot water, after plating with nickel. Bonding of the base to the nickel was assured by heating the control rod to an elevated temperature prior to reactor exposure. Corrosion under reactor conditions was good, but in subsequent use of the alloy, it was encapsulated in stainless-steel tubes so that direct contact with the coolant is eliminated.

### 2.9.3 Rare-Earth Oxides

Several of the rare earths (i.g samarium, europium, gadolinium) have both high thermal-neutron-absorption cross-sections and significant resonances in the epithermal region. The oxides, the only chemical form in which the rare earths have been considered, can be formed into refractory-ceramic pellets. Since the oxides hydrate rapidly in hot water with attendant swelling, their use in water-cooled reactors could lead to difficulties. The oxides may be suitable for use in the gas-cooled reactors. Dispersion of rare-earth oxides in metals such as stainless steel have been prepared. Such dispersions, when suitably clad, can be used for water-cooled reactor-control rods since in the event of clad fracture only the oxide particles on the dispersion surface would get hydrated.

### 2.9.4 Gadolinium Nitrate

The CANDU reactor uses gadolinium nitrate ( $GdNO_3$ ) solution in water as reactor control (shutdown) poison. Tanks full of  $GdNO_3$  concentrated solution is maintained under pressure by helium gas, and is injected into the moderator at a trip signal. This is fast and effective reactor shutdown mechanism. The poison is cleaned from the moderator system by the moderator purification system. Since the poison is introduced to a low-temperature low-pressure moderator, there are no corrosion related concerns.

### 2.9.5 Boron-Containing Materials

The very high cross-section of  $^{10}B$  and the low cost of boron leads to a wide use of boron-containing materials in thermal-reactor control rods and burnable poisons. Unalloyed metallic boron is not suitable for control-rod use.

Boron alloyed or dispersed in stainless steel forms inter-metallic compounds with iron, nickel and chromium of the metal matrix. The result is a major decrease of ductility. Acceptable materials can be obtained by limiting the boron concentration to 2-3 wt. % B. The boron-stainless steel materials have adequate corrosion resistance in water-cooled reactors.

The performance of boron-stainless steel materials is limited because of the  $^{10}B(n, \alpha)$  reaction, which leads to severe swelling localized at the surface of the element (due to short travel distance of the  $\alpha$  particles).

Boron carbide ( $B_4C$ ) is of much greater interest than elemental boron. Boron carbide can be formed into pellets and effective control elements produced by placing these within stainless-steel tubes. In high-temperature gas-cooled reactor designs, control elements can also be produced by dispersing  $B_4C$  in graphite.

In addition to its use in control elements, boron has been used in PWR's for control of reactivity changes with reactor lifetime by dissolving boric acid in the coolant. At the beginning of life, enough boric acid is added to the coolant so that the reactor is just critical with all rods nearly completely withdrawn. As burnup proceeds, the boric acid concentration in the coolant is reduced so as to just maintain criticality ("chemical shim control").

Boron may also be used to compensate for the change in reactivity with lifetime through "burnable poison". In this scheme, a small amount of boron is incorporated in the fuel or special burnable poison rods to reduce the beginning-of-life reactivity. Burnup of the boron causes a reactivity increase which partially compensates for the decrease of reactivity due to fuel burnup and accumulation of fission products. Stainless-steel-boron alloys and dispersions can be used successfully for this purpose since the boron burnup in such rods can be kept low.



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

Characteristic	BWR	PWR(W)	PHWR	HTGR	AGR	LMFBR*
<b>Reference design</b>						
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 <sub>2</sub> O	H <sub>2</sub> O	D <sub>2</sub> O	Graphite	Graphite	—
Neutron energy	Thermal	Thermal	Thermal	Thermal	Thermal	Fast
Fuel production	Converter	Converter	Converter	Converter	Converter	Breeder
Fuel <sup>b</sup>						
<b>Particles</b>						
Geometry	Cylindrical pellet	Cylindrical pellet	Cylindrical pellet	Coated microspheres	Cylindrical pellet	Cylindrical pellet
Dimensions (mm)	10.4D x 10.4H	8.2D x 13.5H	12.2D x 16.4H	400–800 μm D	14.51D x 14.51H	7.0 D
Chemical form	UO <sub>2</sub>	UO <sub>2</sub>	UO <sub>2</sub>	UC/ThO <sub>2</sub>	UO <sub>2</sub>	PuO <sub>2</sub> /UO <sub>2</sub>
Fissile (wt% 1st cofe ave.)	1.7 <sup>235</sup> U	2.6 <sup>235</sup> U	0.711 <sup>235</sup> U	93 <sup>235</sup> U	2.2 <sup>235</sup> U	15–18 <sup>239</sup> Pu
Fertile	<sup>238</sup> U	<sup>238</sup> U	<sup>238</sup> U	Th	<sup>238</sup> U	Depleted U
<b>Pins</b>						
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)	12.27D x 4.1 mH	9.5D x 4 mH	13.1D x 490L	15.7D x 62L	14.89D x 987H	8.65D x 2.7 mH(C) 15.8D x 1.95 mH(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
<b>Assembly</b>						
Geometry <sup>c</sup>	8 x 8 square rod array	17 x 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) <sup>d</sup>	37	271 (C)/91 (BR)
No. fuel rods	62	264	37	132 (SA)/76 (CA) <sup>d</sup>	36	271 (C)/91 (BR)
Outer dimensions (mm)	139	214	102D x 495L	360F x 793H	190.4 (inner)	173F
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].

\*LMFBR-core (C), radial blanket (BR), axial blanket (BA).

<sup>b</sup>Fuel dimensions: diameter (D), height (H), length (L), (across the flats (F), (width of) square (S).

<sup>c</sup>LWRs have utilized a range of number of rods.

<sup>d</sup>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]

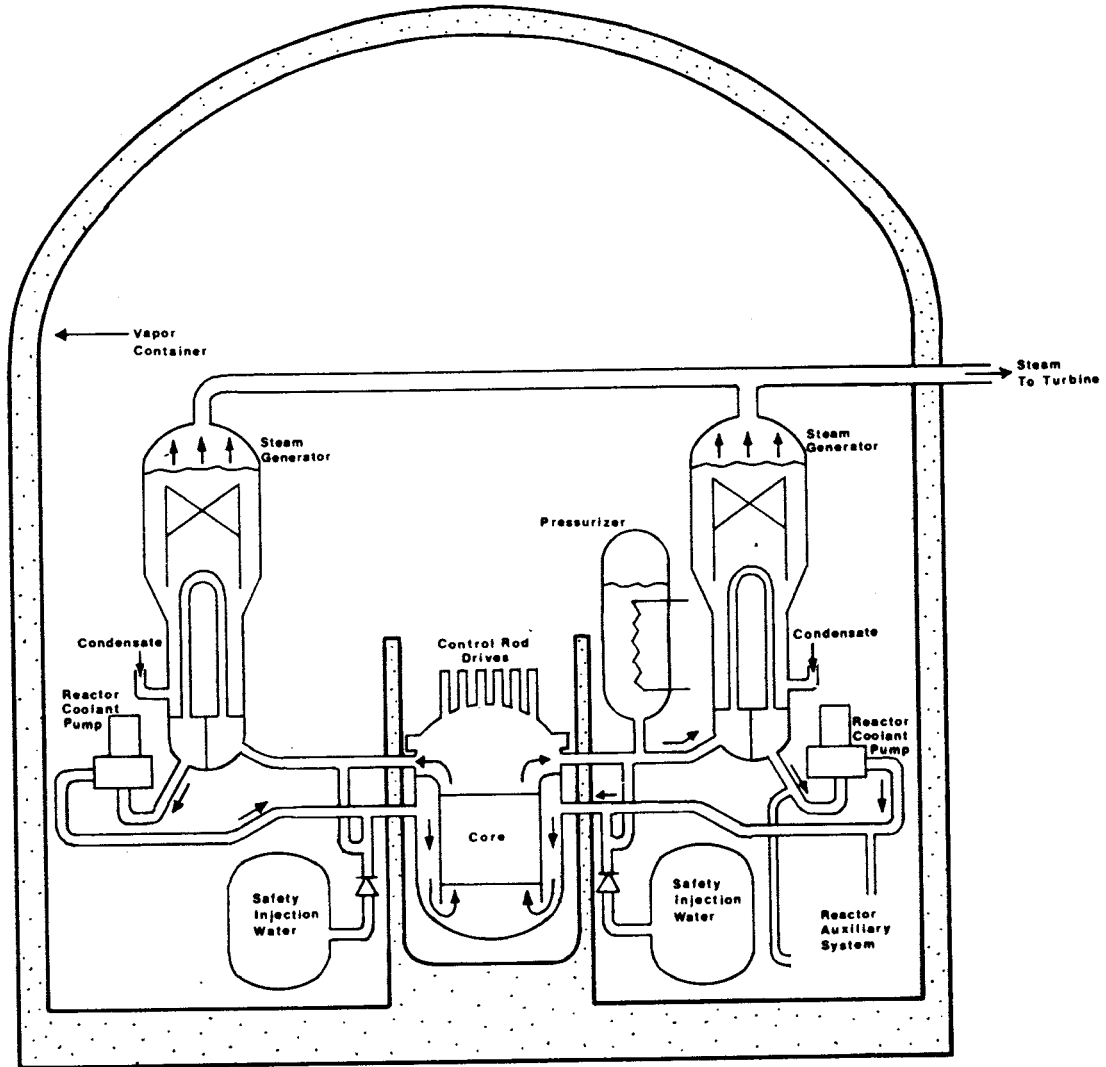


Fig. 1.1(a). Pressurized water reactor system — vessel concept.

Figure 2.11-1: PWR reactor system [WIES77]

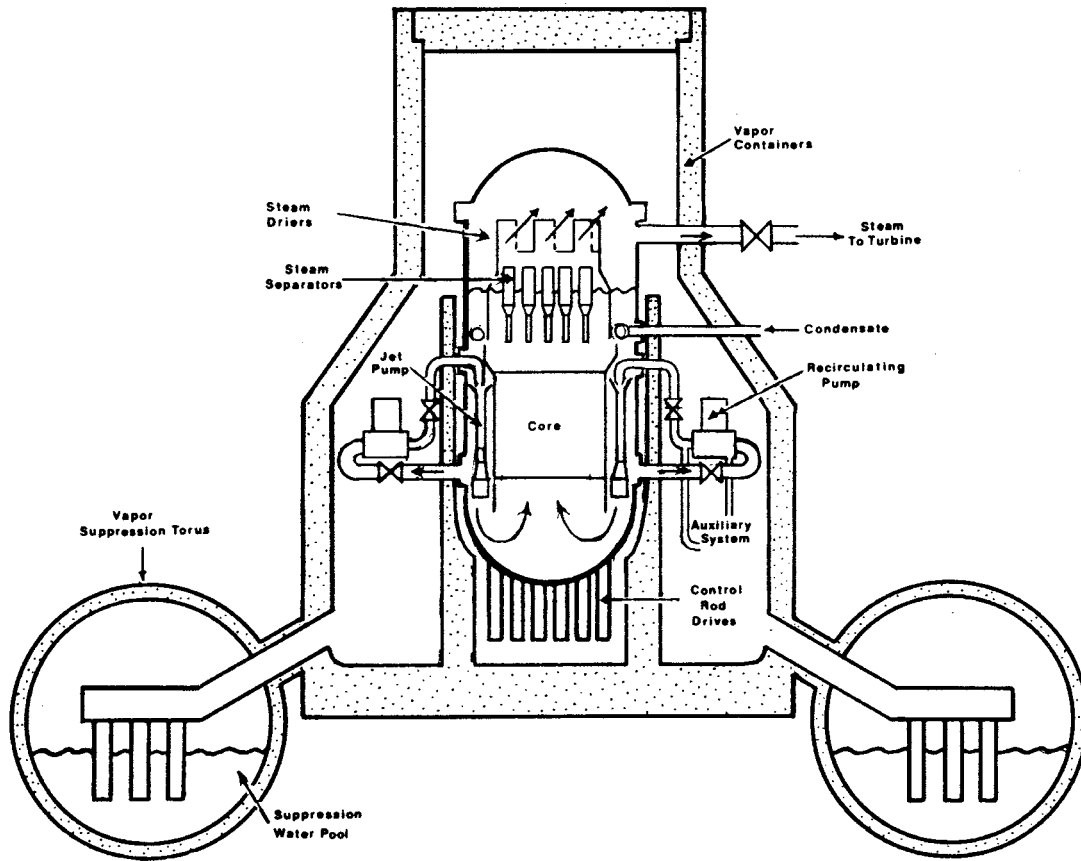


Figure 2.11-2: BWR reactor system

### 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.

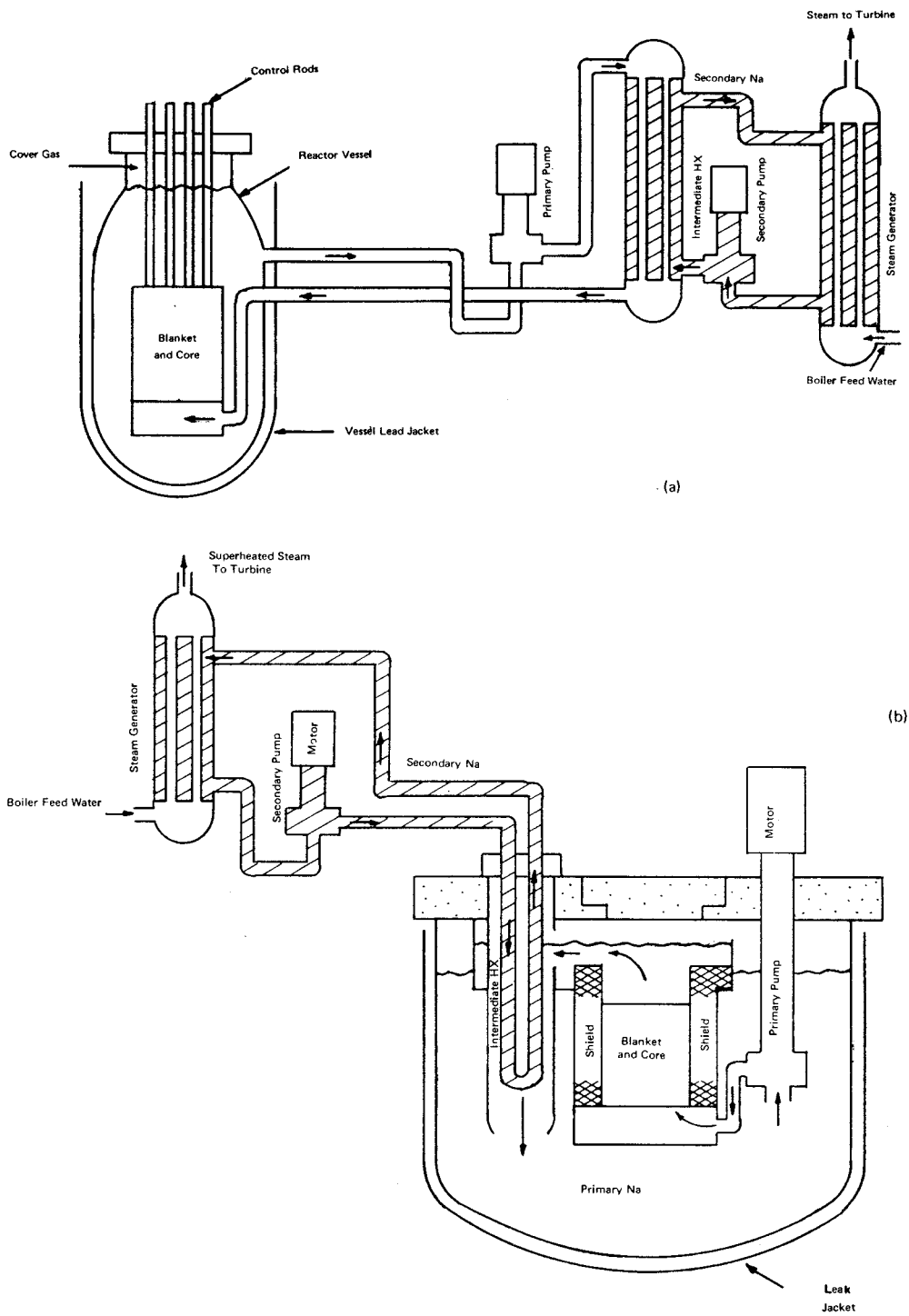


Figure 2.11-3: LMFBR reactor system

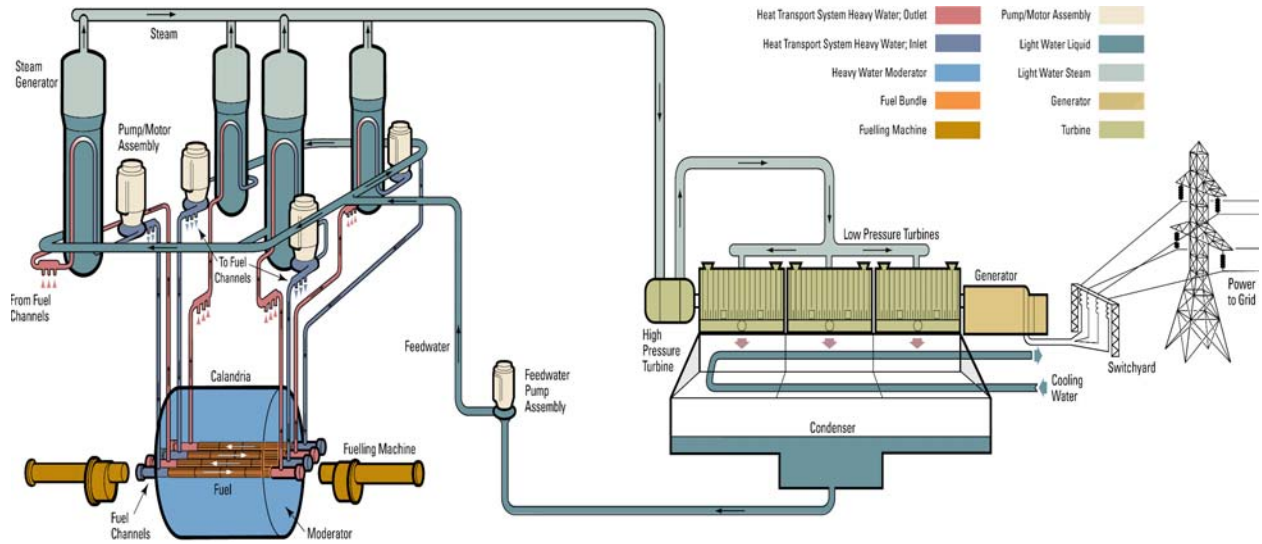


Figure 2.11-4: CANDU-PHW reactor system



### 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  $^{235}\text{U}$  or plutonium-enriched fuel, rather than 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 than 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.

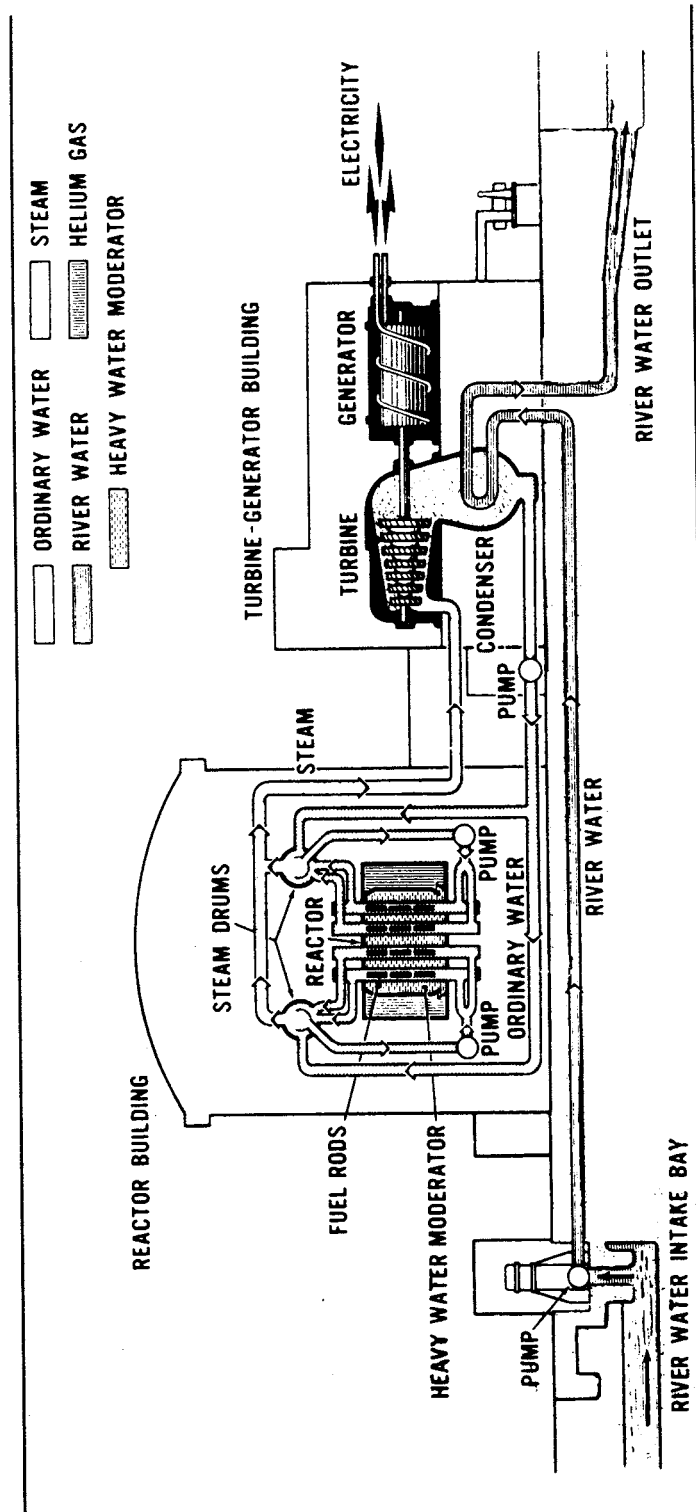


Figure 2.11-6: Schematic illustration of a CANDU-BLW power station

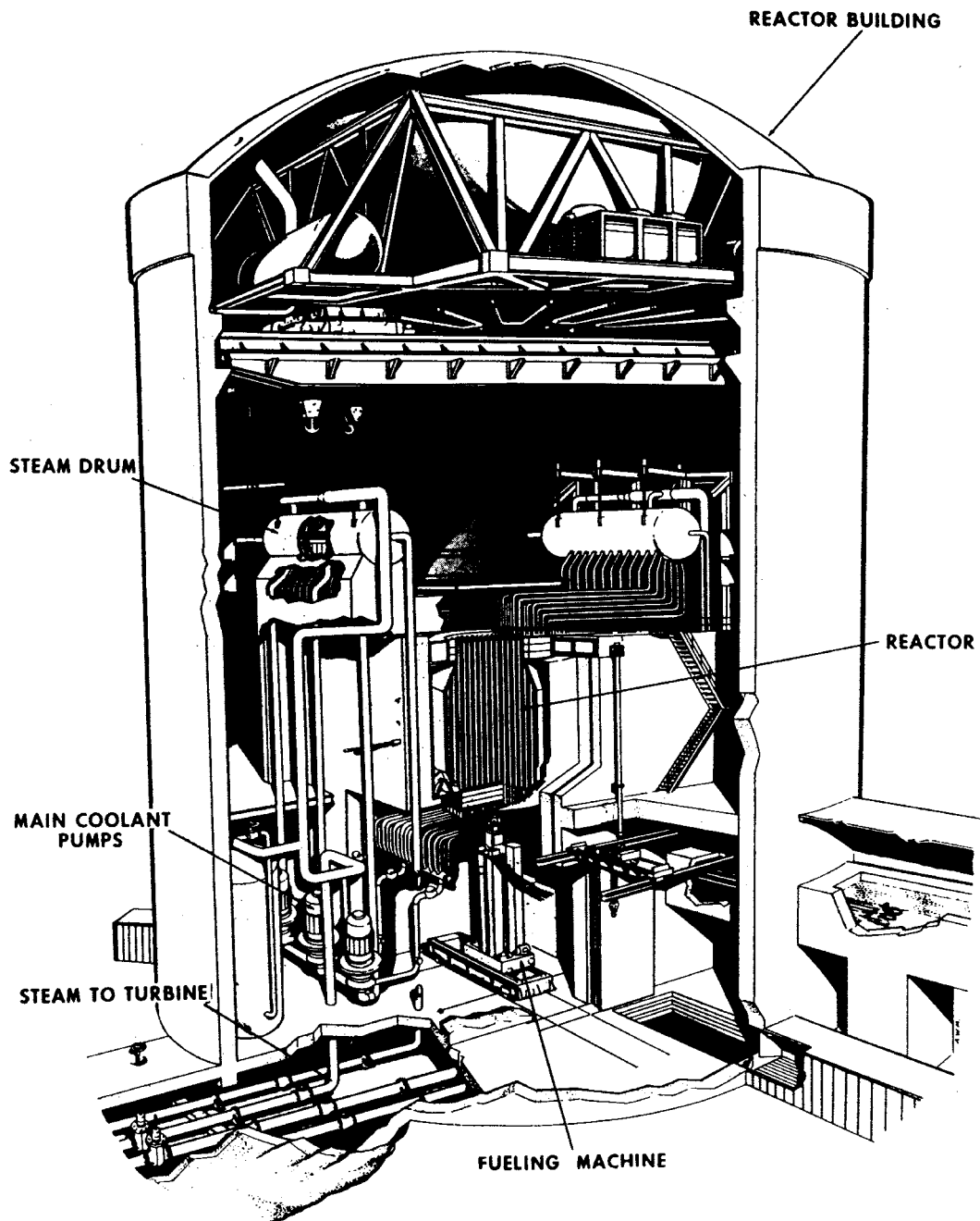


Figure 2.11-6: Cutaway of a CANDU-BLW reactor building

### 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.

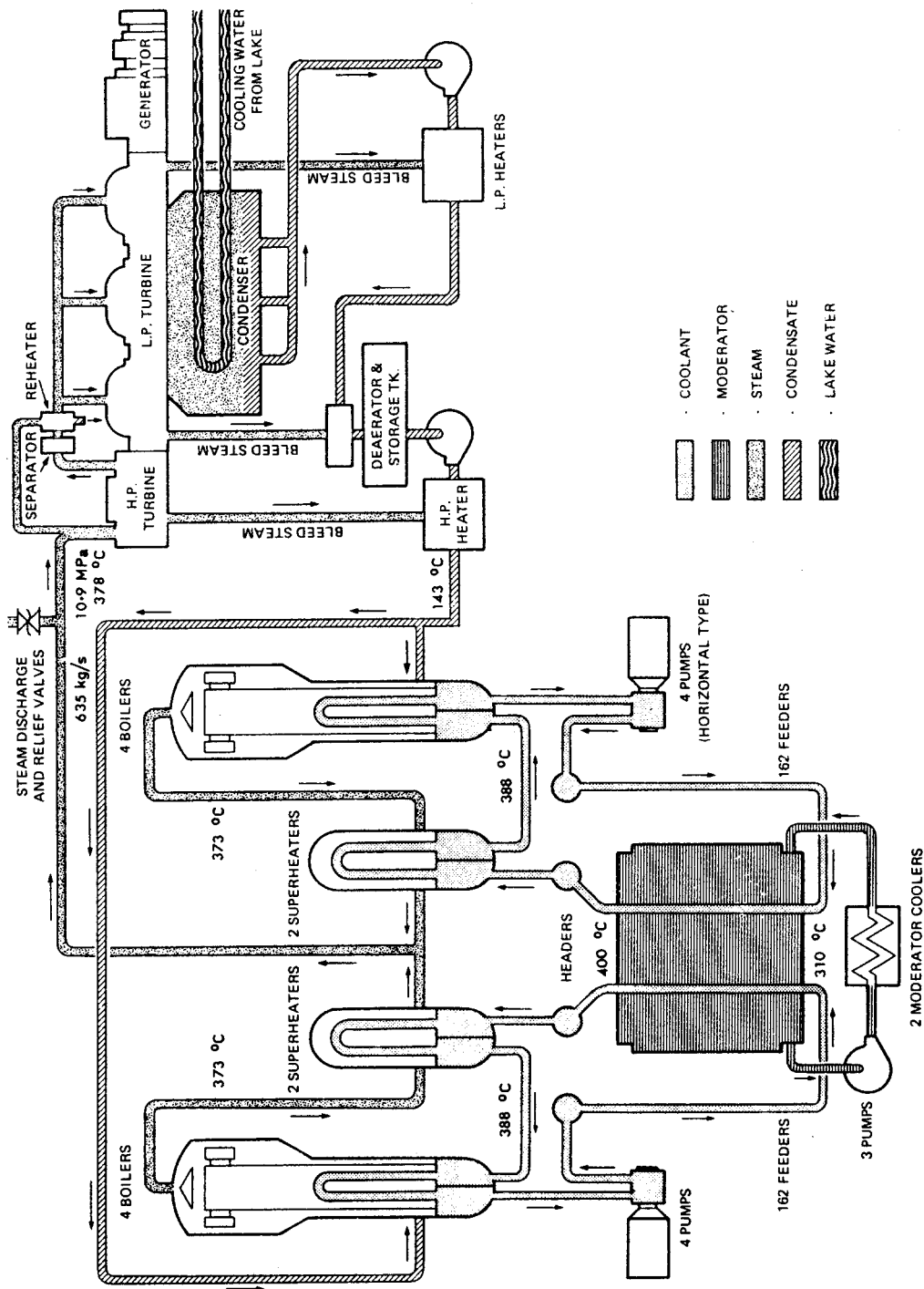
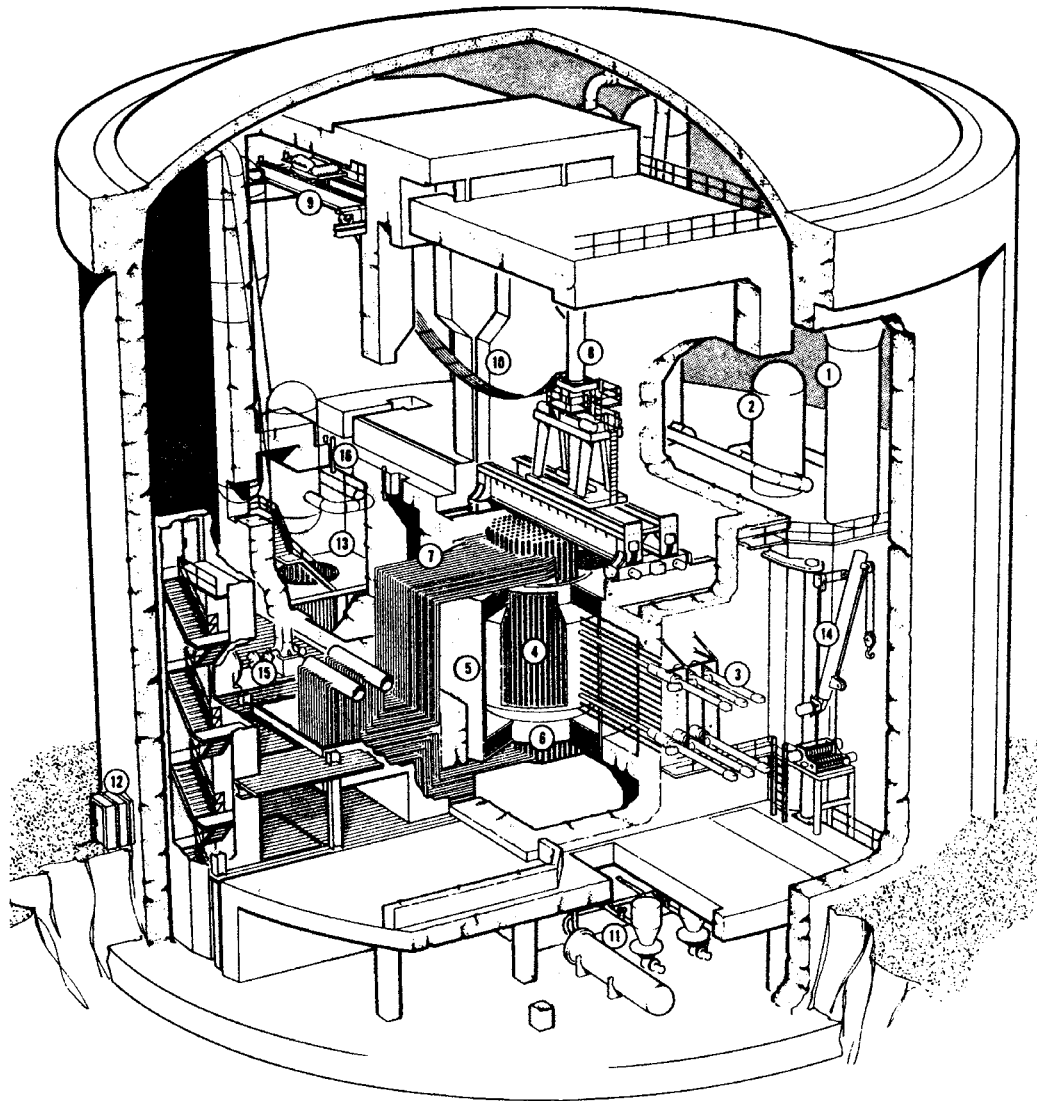


Figure 2.11-7: Schematic diagram of a CANDU-OCR power station



- |                       |                           |
|-----------------------|---------------------------|
| 1. Boilers (8)        | 9. F/M Service Crane      |
| 2. Superheaters (4)   | 10. F/M Vault Door        |
| 3. Booster Rods       | 11. Moderator System      |
| 4. Calandria Assembly | 12. Emergency Airlock     |
| 5. Shield Tank        | 13. Fuel Transfer Bay     |
| 6. End Shield         | 14. Booster Flask Crane   |
| 7. Feeders            | 15. Primary Pumps (4)     |
| 8. Fueling Machine    | 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  $^{235}\text{U}$ .
- 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]