Chapter 2 Design Requirements and Engineering Considerations

[Based on reference BRO72]

2.1 Introduction

All presently developed nuclear power reactors act as sources of thermal energy, producing electricity through the conventional "heat engine" process. This is shown diagrammatically in Figure 2.1. In all current central generating station applications, steam is the final working fluid with more or less conventional steam turbines being employed to drive the electrical generators.

The thermal energy is generated within the nuclear fuel which resides within the nuclear reactor. This thermal energy is transferred from the fuel by a fluid medium called the reactor coolant. This fluid medium may be boiling water, in which case the steam may be used directly in the turbine (the reactor is then called a direct cycle reactor) or it may act as an intermediate heat transport medium, giving up its heat to raise steam in external heat exchangers called boilers or steam generators (the reactor is then called an.indirect cycle reactor).

The various types of power reactors in use today differ regarding the nuclear fuel and the reactor coolants used and also in one further important regard, the type of medium used to slow down or moderate the high energy neutrons produced by the fission process.

We first look at the life cycle of neutrons in the typical nuclear reactor and then consider the various alternative nuclear fuels, coolants, and moderators in current use in commercial power reactors.

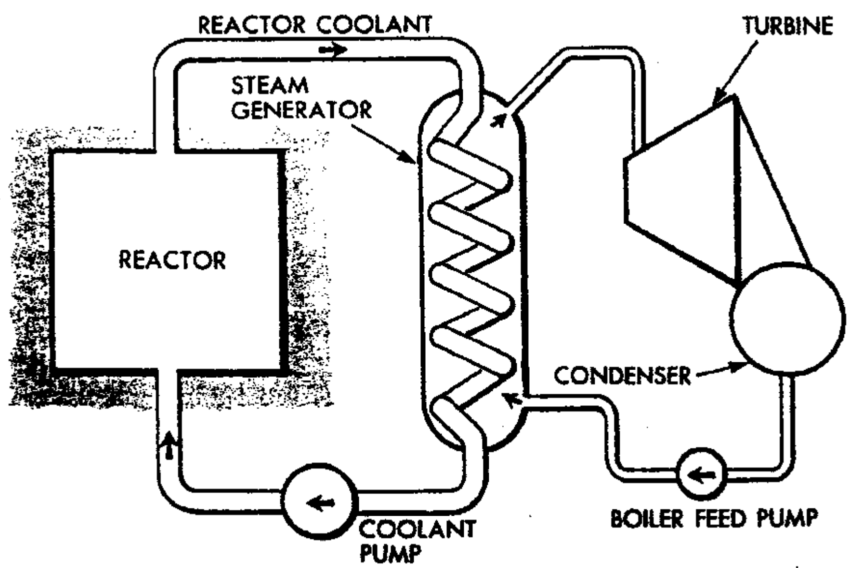

Figure 2.1 Basic power reactor schematic arrangement

2.2 Basic Neutron Cycle

Figure 2.2 depicts the basic neutron cycle wherein a slow neutron is absorbed by a fissile nucleus, causing fission and the emitting of 2 or 3 fast neutrons. The probability of these fast neutrons interacting with other fissile nuclei is low relative to the probability of fission with slow neutrons; hence, the fission neutrons must be slowed down or moderated. This is done by collision with the surrounding media. During the course of this interaction, some neutrons are lost by absorptions that do not lead to fission (parasitic absorptions).

If one thermal (slow) neutron ultimately leads to at least one thermal neutron in the next generation, then a chain reaction is achieved. For this to be the case, the process must exhibit an "economy of neutrons". We need to:

- enhance the probability of neutron moderation
- reduce the probability of neutron absorption
- enhance the probability of fissioning.

This occurs subject to the following constraints:

- safety: the reaction needs to be controllable
- cost: overall cost should be minimized
- process: the reactor system must perform the desired function (ie, generate X MWe) given the limitations such as heat sink capacity, etc.

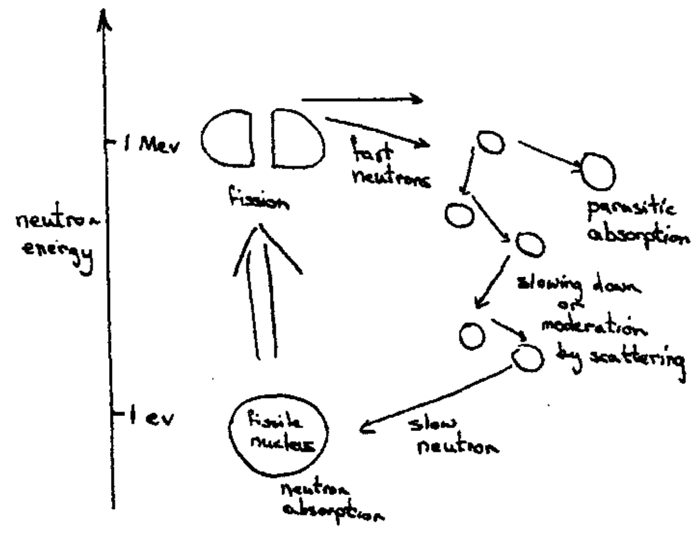

Figure 2.2 The basic neutron cycle

2.3 Possible Fuels

The probability of neutron capture leading to fission (called the fission cross section) is larger for slow neutrons than for fast neutrons. Hence, most practical reactors are "thermal" reactors, that is, they utilize the higher thermal cross sections. Possible fuels include ^{233}U (a fissile material that can be formed from ^{232}Th by neutron bombardment) and ^{239}Pu (also fissile and produced from ^{238}U by neutron bombardment). With one notable exception, all other fissile fuels require a high energy neutron to fission and the cross section is low. The only naturally occuring fuel of significant quantities is ^{235}U, hence most reactors use this fuel.

Naturally occuring uranium is composed of 0.7% ^{235}U. The rest is ^{238}U. This percentage is too low to sustain a chain reaction when combined with most practical moderators. Hence, to achieve criticality, either, the probability of fission must be enhanced or the moderator effectiveness must be enhanced. One group of reactor types (PWR, BWR, HTGR) enrich the fuel (a costly task) and use a cheap moderator (ordinary water or graphite). Alternatively, natural uranium (relatively cheap) is used with an excellent but expensive moderator (heavy water). This is the CANDU approach. In a later section, we shall see why heavy water is such a good moderator.

Enriching the fuel leads to a reactor system with a lower capital cost but higher operating cost than using natural uranium and heavy water. The overall cost over the life if the plant is about the same for either case.

Fast fissions do occur with ^{238}U and can contribute up to 3% to the fission process. But more importantly, some of the ^{238}U is converted to ^{239}Pu which subsequently fissions. In CANDU reactors and other reactors fueled by natural uranium, roughly 50% of the power is generated through ^{238}U. This is less true for reactors with enriched fuel simply because there is relatively less ^{238}U present in the fuel.

2.4 Heat Transfer Considerations

All other things being equal, heat transfer is proportional to surface area. Therefore, best geometries for fuel are those with high area / volume ratios, such as flat plates. However, because a finite thichness of sheath is required, this is not optimum for low parasitic absorption. This is illustrated in figure 2.x.

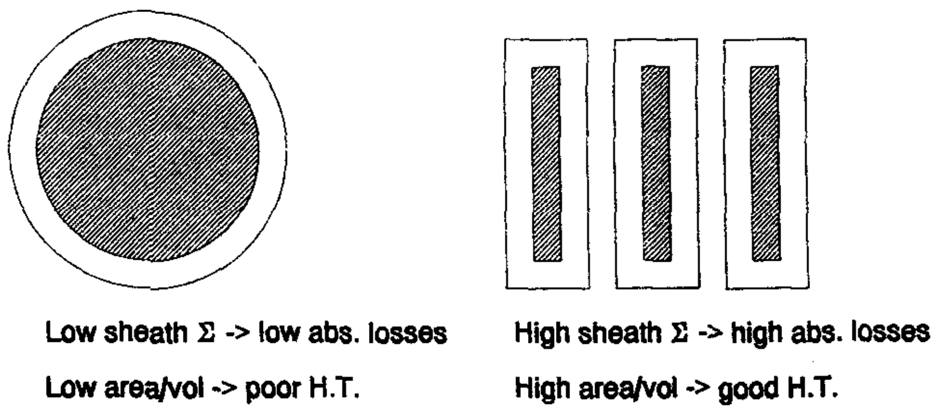

Figure 2.3 Tradeoff between heat transfer and neutron absorption

In addition, to cope with internal pressure generated by fission product gases and swelling at high powers, the circular geometry is better. Tubes are also more economical to manufacture.

Given that many geometries can be made to operate practically and safely, the choice boils down to one of cost.

2.5 Uranium Fuel Forms

In discussing fuel, coolants and moderators, you will note that neutron economy is repeatedly mentioned as an important parameter. This is true even for enriched uranium reactors because the amount of enrichment, and hence the cost of the fuel, is very sensitive to the neutron economy of the reactor. This is particularly so because the enriching of uranium is very costly since it involves an isotope separation process rather than a chemical separation process. No matter which process is chosen, it must utilize the very slight difference in physical properties between the U-238 and U-235 atoms; hence, the process is inherently costly.

In all commercial power reactors, the fuel is used in solid form. Various geometries are employed such as solid rods, plates, spheres, or annular rings. Solid round rods (see Figure 2.4) are used predominantly, primarily because of manufacturing costs. A basic parameter governing fuel design is the external surface area to volume ratio. Good heat transfer to the coolant medium is promoted by high values of this ratio whereas low fuel manufacturing costs and, generally, good neutron economy are promoted by low values of this ratio. This presents a "classical" problem in optimization during the reactor design process, as discussed previously.

In certain power reactors, the fuel material is in the form of uranium metal. Other forms are also used as listed in Table 2.1. Before discussing the merits of the alternative forms, it is useful to consider the desirable properties of fuel material. These are listed in Table 2.2.

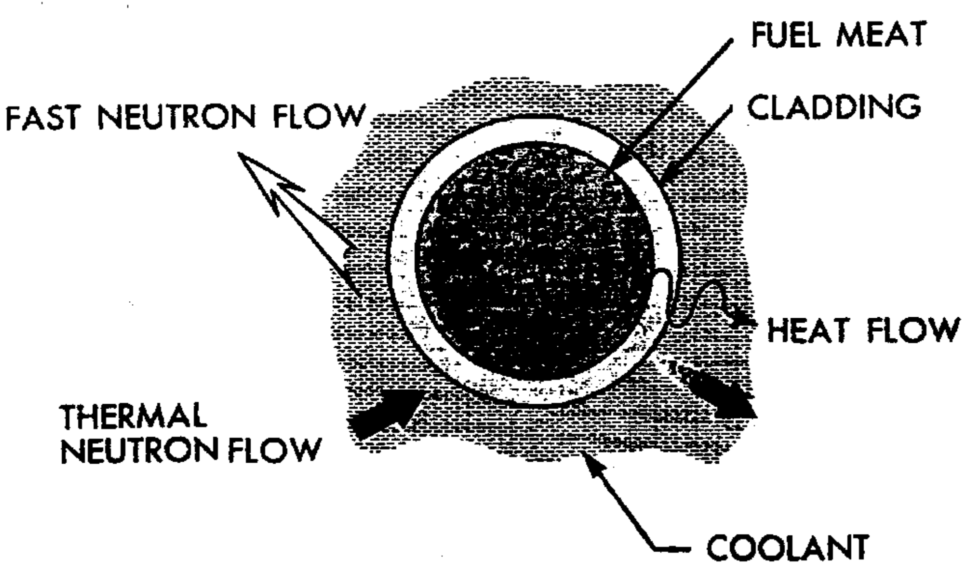

Figure 2.4 Basic reactor fuel arrangement

Table 2.1 Forms of uranium in power reactor fuel

1. URANIUM METAL
2. URANIUM/OTHER METAL. ALLOY
3. CERAMIC URANIUM DIOXIDE
4. URANIUM CARBIDE
5. URANIUM SILICIDE

Table 2.2 Desirable fuel material properties

1. LOW COST - CONSTITUENTS AND FABRICATION
2. GOOD NEUTRON ECONOMY
3. GOOD CORROSION RESISTANCE TO COOLANT
4. PHYSICAL STABILITY UNDER EFFECTS OF IRRADIATION, TEMPERATURE, PRESSURE

Uranium metal is generally lowest in manufacturing cost and highest in neutron economy, the latter ıse of its high density and the absence of the other neutron absorbing elements. On the debit side e ledger, it has poor corrosion resistance to most coolants which is of importance in the event of cladding (to be discussed later) failures. Its geometric stability in reactor use is poor, primarily because of the swelling effects of fission products whose specific volume is, of course, greater than the parent uranium. Small quantities of alloying agents have been found useful but do not fully solve the problem. The problem is aggravated by a metallurgical phase change at relatively moderate temperatures which causes further geometric distortion. This limits the operating power density achievable with the fuel.

Larger quantities of alloying agents such as zirconium can be used which effectively cure the geometric stability problem and the coolant corrosion problem. Unfortunately both the cost and neutron economy suffer. This fuel is used for certain specialized applications where the latter factors are not of overriding importance.

Uranium dioxide is the form in which the uranium fuel is used in the vast majority of today's power reactors. It is somewhat more expensive to manufacture and less neutron economical than uranium metal because of its lower density but possesses excellent corrosion resistance to most coolants and a high degree of geometric stability. Being a ceramic, it is capable of high operating temperatures.

Uranium carbide is attractive as a future fuel for certain types of reactors. It is relatively inexpensive to manufacture (comparable to UO_2) and has somewhat better neutron economy than UO_2 (because of its higher density, but not as good as uranium metal. It has good corrosion resistance to many coolants but unfortunately not to water. Its dimensional stability is good and it can operate at high temperatures.

Uranium silicide is a more recent development having most of the advantages of uranium carbide and, in addition, adequate resistance to corrosion by water coolants.

2.6 Fuel Claddings

In the fission process, new isotopes of a wide variety of elements are produced. These are called fission products. Many of these remain radioactive for significant durations of time after they are generated and, hence, constitute a potential radiation hazard to plant operators and the public at large. It is therefore clearly desirable to keep these fission products "bottled up" within the fuel where they are generated.

This is the primary function of the fuel cladding. This cladding takes the form of an impervious "skin" or "shell" which encloses the fuel material proper. Most cladding materials in current use are metals although ceramic-type materials have had limited use in certain applications. Table 2.3 lists the commonly used power reactor cladding materials. Before discussing the merits and demerits of each it is useful to consider the desirable properties of cladding materials. These are summarized in Table 2.4.

Table 2.3 Alternative fuel cladding materials

1.	ALUMINUM
2.	MAGNESIUM (MAGNOX)
3.	STAINLESSSTEEL
4.	ZIRCONIUM
5.	CERAMICS

Aluminum and its alloys possess many attractive properties such as low cost, easy fabrication, high ductility (important in preventing cladding failures), good neutron economy, and impermeability to fission products. Their major disadvantages for power reactor use are poor mechanical properties at high temperatures and poor high temperature corrosion resistance with most coolants. Since the latter are temperature dependent, aluminum alloys are widely used in research reactor fuels where cladding operating temperatures are low but are not currently used in power reactors.

Magnesium alloys are similar to aluminum alloys in most regards. An alloy called "Magnox" has, however, better high temperature properties and adequate corrosion resistance to permit its use in some CO_2 cooled power reactors.

Stainless steel is a very attractive material in all major regards except for its poor neutron economy. It has been and still is used in a number of enriched uranium reactors where its poor neutron economy is somewhat less important.

Zirconium, in various low-alloy forms, is by far the most common cladding material in current use.

Table 2.4 Desirable cladding properties

1.	CORROSION RESISTANCE TO COOLANT
2.	MECHAN ICAL DURABILITY
3.	HIGH OPERATING TEMPERATURE CAPABILITY
4.	GOOD NEUTRON ECONOMY
5.	LOW COST - BASE MATERIAL & FABRICATION
6.	IMPERMEABILITY TO FISSION PRODUCTS

Despite its relatively high base material cost, it combines to a large degree all of the other desirable cladding properties for use with most coolants.

The use of ceramics and ceramic-type materials have potential for very high temperature applications. Their primary disadvantage is, of course, a lack of ductility which makes them liable to brittle fracture.

2.7 Reactor Coolants

As discussed earlier, the purpose of the reactor coolant is to transport heat generated in the reactor fuel either to the turbine (direct cycle reactor) or to intermediate heat exchangers (indirect cycle reactor). The coolants may be liquids, two-phase liquid/vapour mixtures or gases. Table 2.5 lists the coolants commonly used in current power reactors. Table 2.6 lists the desirable properties of reactor coolants.

Table 2.5 Alternative power reactor coolants

1.	CO_2 GAS
2.	HELIUM
3.	ORDINARY WATER
4.	HEAVY WATER
5.	ORGANIC FLUID
6.	LIQUID METAL

Table 2.6 Desirable features of reactor coolants

1.	HIGH HEAT CAPACITY
2.	GOOD HEAT TRANSFER PROPERTIES
3.	LOW NEUTRON ABSORPTION
4.	LOW NEUTRON ACTIVATION
5.	LOW OPERATING PRESSURE REQUIREMENT AT HIGH OPERATING TEMPERATURES
6.	NON-CORROSIVE TO FUEL CLADDING AND COOLANT SYSTEM
7.	LOW COST

Of the gases, two are in common use: CO_2 and helium. CO_2 has the advantages of low cost, low neutron activation (important in minimizing radiation fields from the coolant system), high allowable operating temperatures, good neutron economy and, for gases, relatively good heat transfer properties at moderate coolant pressures. At very high temperatures, it tends to be corrosive to neutron economical fuel cladding materials and also to the graphite moderator used in most gas-cooled reactors. Its chief drawback, as for all gases, is its poor heat transfer properties relative to liquids. As a result, coolant pumping power requirements tend to be very high, particularly if high reactor power densities are to be achieved (desirable to minimize reactor capital costs).

The other candidate gas, helium, possesses all of the good features of CO_2 and, in addition, is non-corrosive (if pure). Its chief disadvantages are higher costs, particularly operating costs, because helium is very "searching", leading to high system leakage rates unless extreme measures are taken to build and maintain a leak-proof system. This has, however, been successfully done in a number of cases.

Of the candidate liquid coolants, ordinary water is by far the most commonly used. It is inexpensive, has excellent heat transfer properties, and is adequately non-corrosive to zirconium alloys used for fuel cladding and reactor structural components and ferritic or austenitic steel coolant system materials. its disadvantages include only moderate neutron economy and its relatively high vapour pressure at coolant temperatures of interest. It is activated by neutrons in the reactor core but this activity dies away rapidly, permitting reasonable shutdown maintenance access to the coolant system. A further disadvantage is that water transports system corrosion products, permitting them to be activated in the reactor core. These activated corrosion products then create shutdown radiation fields in the coolant system.

The water coolant may be used as a liquid in an indirect cycle system or may be permitted to boil, producing steam in a direct cycle system. Heavy water may also be used as a coolant. Its outstanding advantage is much better neutron economy relative to ordinary water. Its primary disadvantage is its high cost. Otherwise its properties are similar to ordinary water.

Certain organic fluids (primarily hydrogenated polyphenyls) may also be used. They are moderate in cost, have a lower vapour pressure than water, are essentially non-corrosive, and are not significantly subject to neutron activation. Also they do not transport significant quantities of corrosion products which can become activated in the reactor core. Their chief disadvantages include higher neutron absorption than heavy water (but lower than ordinary water), inflammability, and they suffer radio-chemical damage in the reactor core which leads to a requirement for extensive purification facilities and significant coolant make-up costs. On balance, however, they may well see wider application in the future.

Certain liquid metals can be used as coolants. Of these, only sodium and a sodium/potassium eutectic called NaK have achieved significant use. Their advantages include excellent heat transfer properties and very low vapour pressures at high temperatures. Fuel cladding and coolant system materials require careful selection to avoid "corrosion". Their chief disadvantages include incomparability with water (the turbine working fluid), relatively high neutron absorption, a relatively high melting point (leading to coolant system trace heating requirements) and high coolant activation with sustained radiation fields after reactor shutdown.

These disadvantages have effectively precluded the use of liquid metal coolants in commercial power reactors to date with one exception and this is the fast breeder reactor which will be discussed later. In this reactor, the neutrons are "used" at relatively high energy levels where the neutron absorption of the liquid metal is much less, overcoming one of the foregoing disadvantages. In addition, the economics of fast breeder reactors depend on very high core power densities where the excellent heat transfer capability of liquid metals becomes a major advantage. Furthermore, it is desirable in this type of reactor that the coolant not moderate the neutrons excessively. Liquid metals are superior to other liquids in this regard because they do not contain "light" atoms which are inherently effective moderators.

2.8 Neutron Moderators

The most current power reactors are of the thermal type, i. e. , where the energy of the neutrons causing fission is in the thermal range. Since the neutrons produced by the fission process have very high energies, it is necessary that they be slowed down, or "thermalized". The medium employed for this is termed the moderator. It is deployed as a continuous medium surrounding the fuel "cells". The fuel cells form a geometric pattern, termed the reactor "lattice". The optimum spacing between these fuel cells is a function of several variables including the mass of fuel per cell, the mean free path of the neutrons in being thermalized, the degree to which the moderator wastefully absorbs neutrons, the cost of the moderating medium, etc.

The best moderator is something that is the same size as a neutron, ie, the hydrogen atom, ${}^{1}H_{1}$. However, hydrogen does absorb neutrons as well. The deuterium atom, ${}^{2}H_{1}$, at twice the mass of hydrogen, is almost as good a slowing down agent but, since it already has an extra neutron in the nucleus, it has a very low absorption cross section. So, overall, it deuterium is a far better moderator than hydrogen. By using deuterium in the form of heavy water, natural uranium can be used as a fuel. If ordinary water is used, the fuel must be enriched in ${}^{235}U$.

A good moderator has a high scattering cross section, a low absorption cross section and slows down the neutron in the least number of collisions (high lethergy, ξ). Table 2.7 summarizes this. The "figure of merit" is defined as $\xi\Sigma_s / \Sigma_a$.

Before discussing practical moderators, it is firstly useful to consider desirable properties of moderators. These are listed in Table 2.8. Table 2.9 then lists the moderators currently used in commercial power reactors.

Table 2.7 Slowing down parameters of typical moderators [Source: DUD76, table 8-1]

Moderator	A	α	ξ	ρ[g/cm^3]	Number of collisions from 2 MeV to 1 eV	$\xi\Sigma_s$[cm^{-1}]	$\xi\Sigma_s/\Sigma_a$
H	1	0	1	gas	14	—	—
D	2	.111	.725	gas	20	—	—
H_2O	—	—	.920	1.0	16	1.35	71
D_2O	—	—	.509	1.1	29	0.176	5670
He	4	.360	.425	gas	43	1.6×10^{-5}	83
Be	9	.640	.209	1.85	69	0.158	143
C	12	.716	.158	1.60	91	0.060	192
${}^{238}U$	238	.983	.008	19.1	1730	0.003	.0092

Table 2.8 Desirable features of moderator

1.	HIGH MODERATING EFFICIENCY
2.	LOW NEUTRON ABSORPTION
3.	FREEDOM FROM DAMAGE - IRRADIATION, CORROSION
4.	LOW COST - RAW MATERIAL, MANUFACTURE, INSTALLATION

Graphite has been widely used as a moderator for power reactors. The carbon atom is relatively "light", graphite is relatively inexpensive, and carbon is a relatively weak absorber of neutrons. Nevertheless, the carbon atom is sufficiently large, leading to relatively long neutron mean free paths for thermalization, that graphite moderated reactors tend to be large. Furthermore, the relatively large amount of graphite required leads to significant neutron wastage through absorption.

Ordinary water is a much more efficient moderator in terms of the neutron mean free path for thermalization because of its hydrogen atoms. It is also very inexpensive. Unfortunately, however, hydrogen also has a significant "appetite" for absorbing thermal neutrons which hurts neutron economy.

Heavy water is almost as good as ordinary water in terms of neutron mean free path since the deuterium atoms (which replace the hydrogen atoms in ordinary water) are relatively "light". Its outstanding advantage, relative to ordinary water, is that it has a very small "appetite" for absorbing neutrons. Hence, it promotes a high level of neutron economy. Its major disadvantage is its high cost.

Table 2.9 Alternative power reactor moderators

1.	GRAPHITE
2.	ORDINARY WATER
3.	HEAVY WATER

2.9 Moderating Arrangements

How do the fuel, the coolant, and the moderator "fit" together to form practical power reactors? The currently established alternatives are shown in Figure 2.5. If ordinary water is used as both coolant and moderator, it is practical to arrange the fuel "rods" in cluster assemblies as shown. The clusters abut against each other. The space between the individual fuel rods is occupied by ordinary water which acts as both moderator and coolant. A relatively small volume of water is required because of the very short neutron mean free path with a hydrogen-based moderator. Hence, the fuel rods can be located relatively close to each other. This arrangement is used in both PWRs and BWRs.

If graphite (a solid) is used as the moderator, it is possible to arrange the graphite and fuel into abutting composite assemblies.

Coolant passages are arranged through the fuel rods (annular form) or through the graphite. The former approach is used in one Russian reactor type where the coolant is water and steam (for superheating). The latter is used in HTGCR's where the coolant is helium and the fuel is uranium carbide, permitting extremely high fuel operating temperatures.

A third arrangement is where the fuel is in the form of assemblies completely separated from the moderator. This arrangement is used in heavy water moderated and most graphite moderated reactors.

T . e choice between these alternatives is influenced by many factors, both of a neutron physics nature and a practical engineering nature, and is very dependent on the particular choice of fuel, coolant and moderator.

Time does not permit a detailed discussion of all of these, although many of the factors have been touched on in a qualitative way in the preceding sections. Most of the rest, also in a qualitative way, will be touched on in the next section which deals with specific power reactor types.

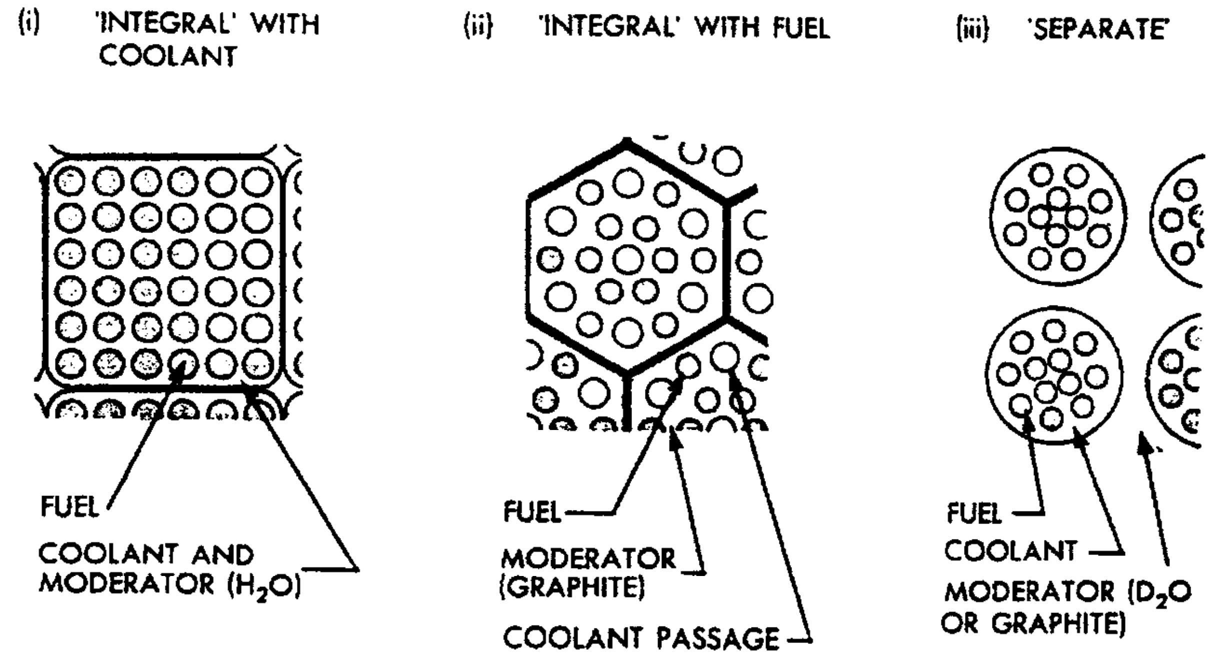

Figure 2.5 Moderating arangement

2.10 HTS Design Requirements and Engineering Considerations

This section introduces the heat transport system and associated systems by a discussion of design requirements and engineering considerations which guide the design of systems to transfer fission heat to the coolant for the production of steam.

The fissioning process results in heat generation in the nuclear fuel and surrounding media. This thermal energy can be utilized to produce electricity or process steam by the use of a heat transport medium, the coolant. Here we will discuss some of the thermalhydraulic features which characterize the CANDU system, but the story is similar for PWRs..

The main objectives of the heat transport system are to provide heat transfer at high thermal efficiency and to allow the maximum amount of energy to be extracted from the fuel without surpassing safe limits.

The requirements for such a system can be summarized as follows:

a) Due to the decay heat produced by the fuel even when the reactor is shut-down, continuous coolant flow must be provided. This leads to the requirement for pumps, pump flywheels, standby cooling, thermosyphoning, etc.

b) Costs should be minimized with due regard for the other requirements. This usually leads to trade offs between, for example, heavy water (D_2O) costs, pumping power costs, equipment and piping size and costs, layout and engineering constraints.

c) Layout should minimize man-rem exposure and maximize maintainability and accessibility within the constraints of other considerations.

d) Provision must be made for pressure and inventory control of the heat transfer system. Excessively high pressure could damage the fluid boundaries (pipes, etc.). Low pressure could lead to high coolant voiding and possible fuel damage and to pump damage from cavitation. Low inventory jeopardizes coolant circulation and pressure control.

e) The system must be sufficiently reliable since downtime leads to high replacement energy costs, high man-rem exposure and repair costs.

f) The design should provide high process efficiency.

g) The system should exhibit ease of constructibility to reduce initial costs and time of construction, and to enhance maintainability.

h) The system should meet and, preferably surpass all safety and licensing requirements.

Various coolants can be used in the CANDU design to achieve the above objectives and requirements.

Any nuclear station design employs a tradeoff in design features to best achieve the lowest cost power within the safety limits. The U.S. nuclear industry, for instance, because of the availability of enriched uranium from existing UF_6 diffusion plants, chose to use enriched uranium and H_2O coolant in order to achieve the necessary neutron economy.

From a neutron economy viewpoint, the medium surrounding the fuel, ie., the coolant and the moderator, must not absorb neutrons and must moderate the neutron energy by a minimum of collision interactions. D_2O is by far the best moderator/coolant from this viewpoint. The cost, however, is high at approximately $300/kg in 1980 dollars.

Using H_2O as the coolant, as in the CANDU-BLW, Gentilly-1, gives poorer neutron economy than the CANDU-PHW and requires booster rods for startup until the positive void coefficient of reactivity adds a sufficient positive reactivity to maintain criticality. Because of this and because of reactivity control difficulties associated with the large void coefficient of reactivity, no new commercial CANDU-BLW's are planned. Organic coolant, Monsanto OS-84, requires slightly enriched fuel (1.2 to 2.4 wt%). This option was found feasible but, due to the success of the CANDU-PHW, no commercial OCR's are planned.

Another nuclear consideration is that the coolant should have a low induced radioactivity. Both H_2O and D_2O produce N-16 and 0-19 which emit γ's in the 6-7 MeV range. This leads to reduced accessibility and maintainability while on power. the short half life (<1 minute) allows shutdown accessibility. Tritium, H^3 or T, has a 12 year half life and represents a major dose commitment for the station. Since tritium is a β emitter, the problem is one of leakage, leading to possible absorption/ingestion by humans. Organic coolant has very little induced reactivity and aids in ease of operations, accessibility, etc.

The coolants should also be stable in a radiation environment. At the high system pressure of the heat transport systems of H_2O and D_2O, radiolysis is not a problem. However, since hydrogen and deuterium have a tendency to diffuse through the pipework, the heat transport system becomes concentrated in oxygen and enhances corrosion. Supplying an excess of hydrogen or deuterium prevents this occurrence by driving the chemical equilibrium balance towards the associated state.

Organic coolant is more susceptible to radiolysis and requires degassing and makeup.

The choice of coolant also depends on other factors, such as pumping power, heat capacity, heat transfer coefficients, flowrates, pressure drop, boiling point, freezing point, corrosion, flammability, thermal stability, and cost.

Water (both D_2O and H_2O) is an attractive heat transport fluid since it offers a good balance of the above considerations. The specific heat, density and thermal conductivities are high compared to alternatives such as N_2, CO_2 and OS-84 (organic). Since pumping power is given by:

Pumping power = pressure drop x volumetric flow rate,

water requires less pumping power for a given heat removal.

For the Bruce reactors (which generate about 750 MWe), approximately 24 MW's of pumping power are required for each reactor. Of this 24 MW, roughly 90% (or 21.5 MW) ends up in the primary heat transport system as heat due to friction. At an overall station efficiency of 30%, the net unit load for pumping power is 24 - 21.5 MW (bearing and windage losses) plus 21.5 x .7 = 15 MW (rejected energy) for a total of 18.5 MW. This represents over 2% of the electrical power generated. Since MW saved here by reducing pumping power is gained as electrical output, considerable emphasis is placed on lowering pumping power.

Limiting flowrates for water depend on many factors such as temperature, the presence of boiling, water chemistry, geometry and flow regime. Fretting considerations have led to a 10 m/sec limit on fuel channel velocity in single phase water. Erosion/corrosion considerations have led to 4.3 to 6.1 m/s (14 to 20 ft/s) in the steam generator tubes and 16.8 m/s (55 ft/s) in heat transport piping. These limits may change as more is learned about the limiting phenomena.

The fuel distribution in the coolant is such to maximize the surface to volume ratio of the fuel so that the highest heat transfer surface can be exposed to the coolant for maximum heat transfer without drying out the fuel surface. However, if carried to extremes the fuel volume in the core will be lower than optimum and parasitic neutron absorption due to the sheath will increase. Present designs employ 37 or 28 elements in a fuel bundle.

The use of boiling in the coolant permits higher heat transfer due to the high heat transfer coefficient of post-nucleate boiling.

Ideally, the coolant temperature should be as high as possible for maximum overall thermal efficiency. Thus a high boiling point, low vapour pressure liquid is desirable so that the heat transport system can be at the lowest possible pressure. This reduces the thickness of the pressure boundary and thus is important for reducing the parasitic burnup in the core. Organic coolant is far superior to water from this point of view.

For the case of organic coolant, the secondary side H_2O pressure is higher than the primary side OS-84 pressure. Thus boiler tube leaks will cause a water leak into the primary coolant system.

Freezing point concerns for H_2O and D_2O are minor. For OS-84 provision must be made to prevent freezing while shutdown and cold. Continuous coolant makeup reduces this problem.

Corrosion of the heat transport system materials must be minimized because of possible deterioration, flow restrictions and contamination with active isotopes.

The CANDU-PHW heat transport system has water coolant, low cobalt carbon steel piping, stainless steel end fittings, zircalloy pressure tubes and Monel or Incoly steam generator tubes. A pH of 10.2 to 10.8 is maintained by lithium hydroxide. Hydrogen gas is added to keep the dissolved oxygen content low to help minimize corrosion. The intent is to produce and maintain a continuous and adherent film of magnetite on all the carbon steel surfaces. Corrosion with organic coolant is a lesser problem, controlled by degassing, by using N_2 cover gas, and by a dechlorinator system.

No flammability or thermal stability problems exist with water (except for the possible Zr-water reaction producing H_2 during a LOCA giving the potential for H_2 explosion) but organic coolant is combustible, although it will not sustain combustion on its own. Organic coolant is also not as thermally stable as water.

The current cost of D_2O (\$300/kg - 1995 dollars) is high, making it the more expensive coolant. This contributes to a high capital cost for the CANDU-PHW but a low operating cost due to the efficient use of natural U.

2.11 Power Reactor Types

It is not much of an exaggeration to state that in the early days of power reactor development there were champions for every possible combination of the fuels, coolants, moderators, and moderator arrangements discussed in the preceding sections and a few more besides. Many of these have fallen by the wayside, either because of basic, inherent shortcomings or, in some cases, because their champions could not rally adequate support. This is, of course, natural with an emerging technology. A number of the possible combinations have reached the point of commercial exploitation. These are described briefly in the following subsections.

2.11.1 "Magnox" Reactors

These are graphite moderated, CO_2 gas cooled reactors fuelled with natural uranium metal clad with a magnesium alloy called Magnox. They have derived their generic name from this latter feature. Figure 2.6 shows a schematic arrangement of one version of this reactor type.

This type of reactor was pioneered by the British and French and was a natural outgrowth of earlier air-cooled, graphite-moderated research and plutonium production reactors. A significant number were built in Britain and France with a few exported to other countries. Early versions used steel reactor pressure vessels with external heat exchangers (boilers) and gas circulating blowers. Later versions, as per Figure 2.6, employed prestressed concrete pressure vessels incorporating the reactor core, heat exchangers and coolant circulation blowers. This was primarily a cost reduction measure, although safety advantages in terms of risk of coolant system rupture were also claimed (likely valid).

Primarily because of coolant temperature limitations imposed by the uranium metal fuel and the Magnox cladding, only relatively modest turbine steam conditions are achievable, limiting the station overall efficiency to ~ 30%.

As is typical of all natural uranium power reactors, the Magnox reactors are fuelled on-load. This is because large quantities of excess reactivity, in the form of additional U-235, is not "built into" the new fuel.

The in-service availability of the Magnox reactors has proven to be relatively good. On-load refuelling helps in this regard. Nevertheless, their relatively high capital cost and relatively modest achievable fuel utilization has led to the discontinuation of construction of further reactors of this type.

2.11.2 AGR

The AGR (advanced gas cooled reactor) has been developed in the U. K. as a successor to their Magnox line of reactors. Several are now under construction. They differ from the latest Magnox reactors primarily in the fuel used. The fuel is UO_2 clad in stainless steel. This permits rather higher fuel temperatures and, hence, coolant temperatures to be achieved, leading to conventional fossil fuel steam conditions (2400 psi, 1025°F/1025°F). The fuel is in the form of a cluster of small diameter rods, permitting relatively high power levels to be achieved. This reduces the size of the reactor core

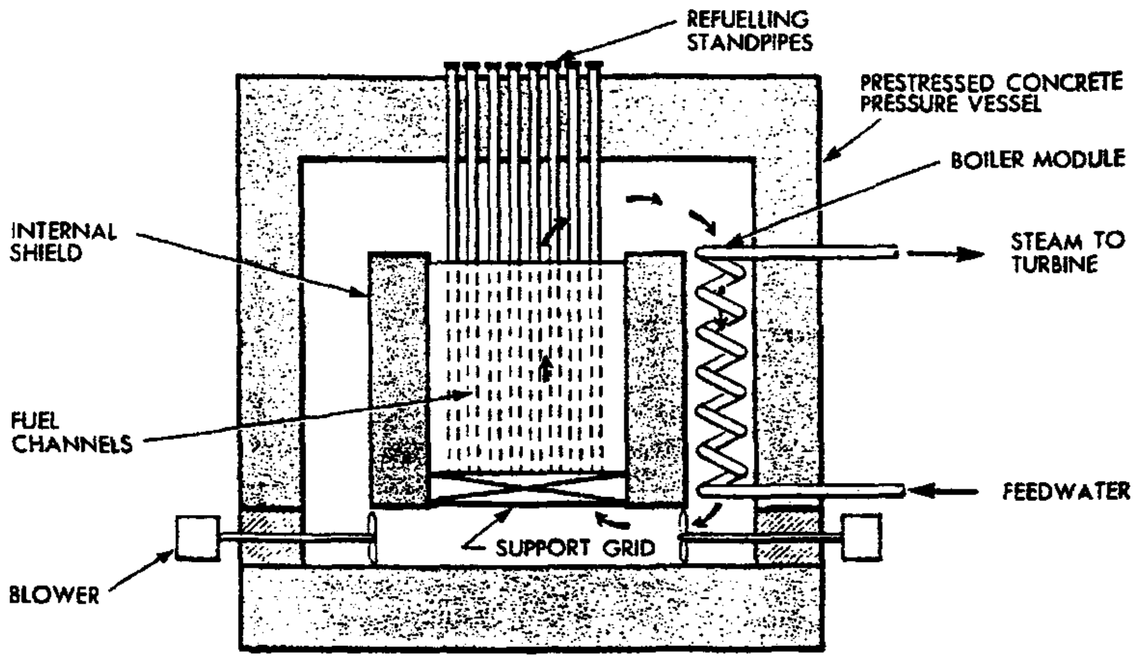

Figure 2.6 Schematic arangement - Gas Cooled Reactors

relative to the Magnox reactors where the fuel is in the form of large single elements. However, because of these fuel changes, the AGR requires some fuel enrichment.

Figure 2.6 also applies to the AGR reactor type.

The British currently appear to have decided that the AGR is not fully competitive with some other types of power reactors. Hence, this design, like the Magnox type, appears to be "dead-ended".

2.11.3 HTGCR

This type represents the next evolutionary step in the Magnox-AGR line of gas-cooled, graphite-moderated reactors. It is being developed by Gulf General Atomic in the U.S. and by the West Germans and British.

The HTGCR differs from the AGR in two major respects. The first is the use of helium as the coolant in place of CO_2. This permits even higher coolant temperatures without inducing a chemical reaction with the graphite moderator. The second relates to the fuel. The fuel uses fully enriched (93%) U-235 mixed with thorium. Thorium absorbs neutrons and is converted, after a radioactive decay chain, to U-233 which is fissile. As a result, the reactivity of the fuel remains high even after very long irradiation, the U-233 replacing the U-235 as the latter is burned up. Their fuel is in the form of mixed carbides. It is manufactured in very small spheres which are coated with pyrolytic graphite, the latter providing the cladding. These spheres are compacted into holes in large graphite assemblies, forming an integral fuel and moderator assembly as per Figure 2.5.

The very high achievable coolant temperatures lead to high steam cycle efficiencies, or alternatively, make possible the ultimate use of gas turbines directly driven by the coolant. Figure 2.6 applies to the former approach since the basic system is the same as for the Magnox and AGR concepts.

The fuel probably represents the major development problem yet to be completely solved in terms of achieving attractive long-term fuelling costs. This reactor type, because of its high thermal efficiency, would see some preference in areas where waste heat rejection presents a particular problem. The development of the direct cycle gas turbine version would be particularly attractive in this regard.

2.11.4 PWR

The PWR (pressurized water reactor) has, to date, been the world's most widely accepted power reactor type. It got its start in the development of the PWR propulsion reactors for the U. S. nuclear submarines.

In this type of reactor, ordinary water is used both as the coolant and the moderator. The fuel is in the form of clusters of enriched UO_2 rods clad in zirconium alloy or, in some cases, austenitic stainless steel. These clusters are square in shape, i.e., the rods form a square array in each cluster assembly, with the clusters, in turn, being closely packed in a square array forming the reactor core, see Figure 2.5. As is shown in Figure 2.7, the reactor core is located in a large steel pressure vessel. .he water coolant at high pressure (~ 2000 psi) is circulated by external pumps into the reactor vessel, flows upwards through the fuel clusters, out of the vessel to heat exchangers, and from the heat exchangers back to the pumps. On the secondary side of the heat exchangers, water is boiled forming saturated steam which drives the turbine. This steam is generated at ~ 750 psi, leading to a relatively low overall station efficiency (~ 30%).

In order to refuel the reactor, it must be shut down, cooled out and depressurized. The top of the pressure vessel is then removed and the fuel assemblies changed. This refuelling is normally done annually. In order to operate for long periods without refuelling, the new fuel is relatively highly enriched in U-235.

While the fuel is new, the excess reactivity in the core is compensated for by a neutron poison dissolved in the coolant/moderator water. As the fuel burns up, the poison is gradually removed by ion exchange columns.

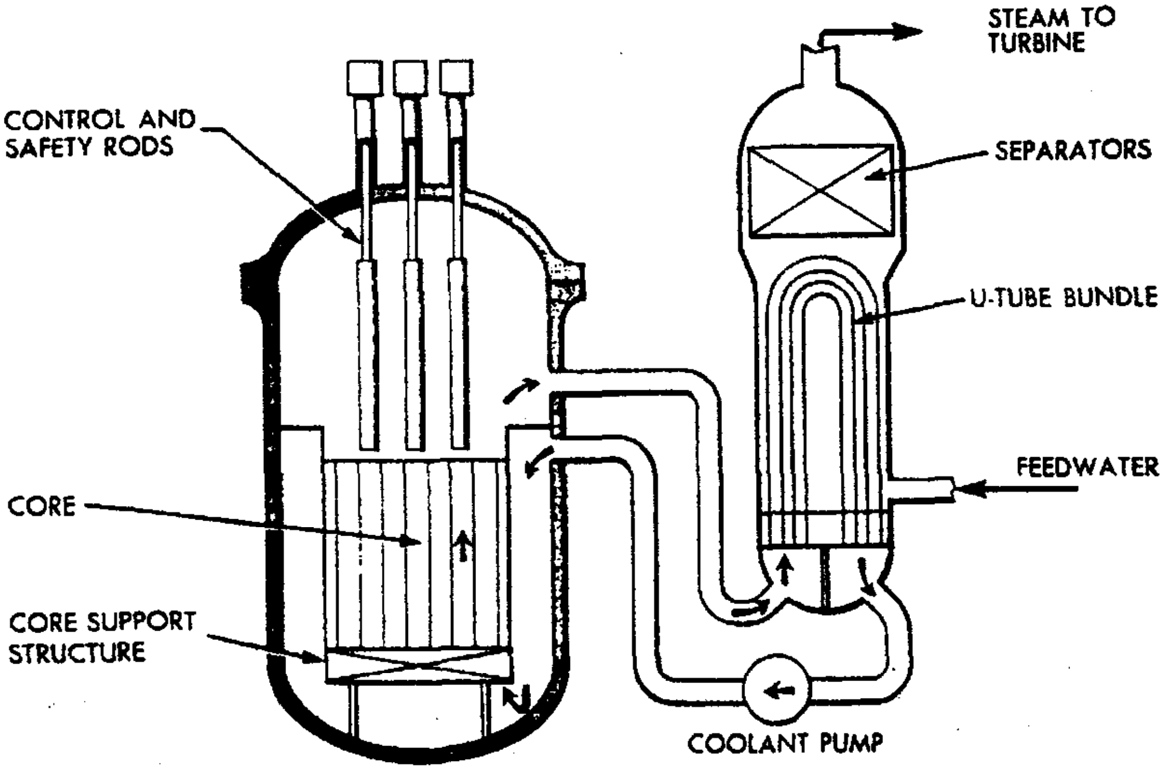

Figure 2.7 Schematic arrangement PWR

2.11.5 BWR

The BWR (boiling water reactor) is second only to the PWR in terms of world-wide acceptance. It is similar in many respects to the PWR, the basic difference being that the light water coolant is permitted to boil in the reactor core. The steam thus produced is separated from the coolant water by centrifugal separators located in the reactor vessel above the core and fed directly to the turbine at ~1000 psi pressure. The general arrangement is as shown in Figure 2.8.

With this arrangement, the turbine plant is "active" because of activity induced in the reactor coolant (primarily N-16). As a result, the turbine plant is more or less inaccessible during operation; fortunately, however, this activity dies out quickly following shutdown, permitting normal direct access maintenance.

While the BWR appears simpler than the PWR, it has not been able to secure a clear economic advantage over the latter. The two types have run "neck and neck" in the acceptance race for years, both in the U.S. and in many other countries.

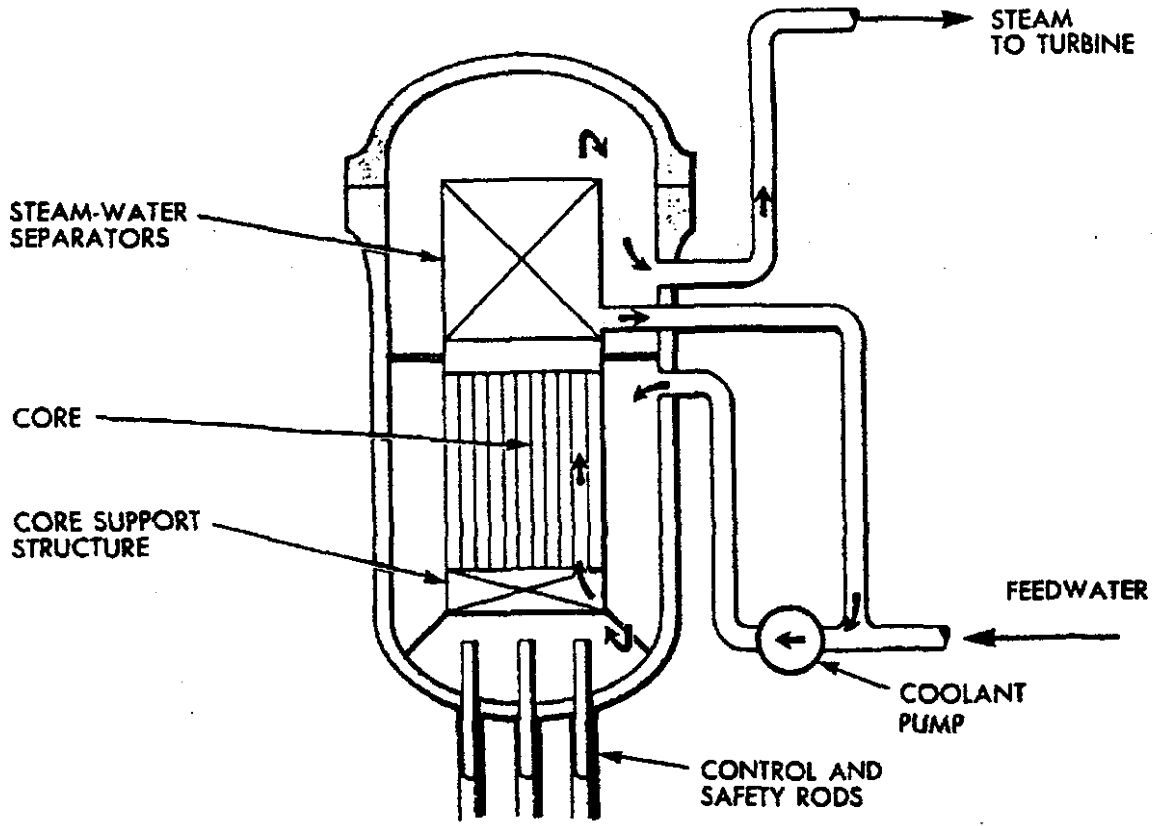

Figure 2.8 Schematic arrangement BWR

2.11.6 LMFBR

Before discussing the last major current commercial type of power reactor, I would like to briefly describe the liquid metal cooled fast breeder reactor (LMFBR). While no reactors of this type are currently in commercial operation.

Firstly, a few "basics". All of the previously described reactors are of the thermal type, i. e. , the fissions in the fuel are primarily induced by thermal neutrons. It is, however, possible to sustain a chain reaction with high energy, i.e., fission, neutrons, provided the fuel is highly enriched with fissile material such as U-235 or Pu-239. Furthermore, an average of rather more than two neutrons are born from each fission. One of these neutrons is required to induce the next fission, leaving a surplus of rather more than one neutron which can be absorbed by a "fertile" material such as U-238, producing fissile Pu-239. We can produce fissile material as rapidly as we use it up. This is called "breeding". In fact, it is possible to produce more fissile material than is used because the average number of neutrons produced per fission is >2. The excess is referred to as the "breeding gain". Clearly this can only be done if the neutron economy is high, i. e. , relatively few neutrons are wasted.

This possibility of breeding is very attractive as a means of extending the power available from uranium since, as you will remember, less than 1% of natural uranium is fissile. If a substantial part of the other ~99% can be converted to fissile Pu-239 as a byproduct of reactor operation, then the world's uranium reserves can be stretched enormously. The fast breeder reactor is one way of doing this; hence, the widespread interest in this concept.

The basic arrangement of a liquid metal cooled fast breeder reactor is shown in Figure 2.9. The reactor core consists of a closely packed array of highly enriched (U-235 or PU-239) oxide rods clad in a high temperature resistant metal. This core is surrounded on all sides by a "breeder blanket" of fertile U-238 (also in clad oxide form) rods. The excess fission neutrons produced in the core "leak" out of the core and are absorbed in the blanket rods. Both the core and blanket are cooled by a flow of liquid sodium. This sodium is, in turn, cooled in heat exchangers and returned to the reactor by more or less conventional centrifugal pumps. The heat exchangers are cooled by a second flow of sodium which, in turn, is cooled in a second set of heat exchangers which produce steam for the turbine. The purpose of this intermediate sodium loop is to provide completely positive isolation between the sodium cooling the reactor and the turbine cycle steam and water, thereby ensuring that an inleakage of water cannot contaminate the reactor coolant.

Despite this intermediate loop, the reactor operating temperature is sufficiently high to permit steam to be produced at modern fossil fuelled plant conditions (~2400 psig and 1000°F with single reheat to 1000°F).

Because of the complex economies and technical problems associated with the LMFBR, while many people feel this is the "reactor of the future", it is probable that the future in this case will be quite a few years in coming.

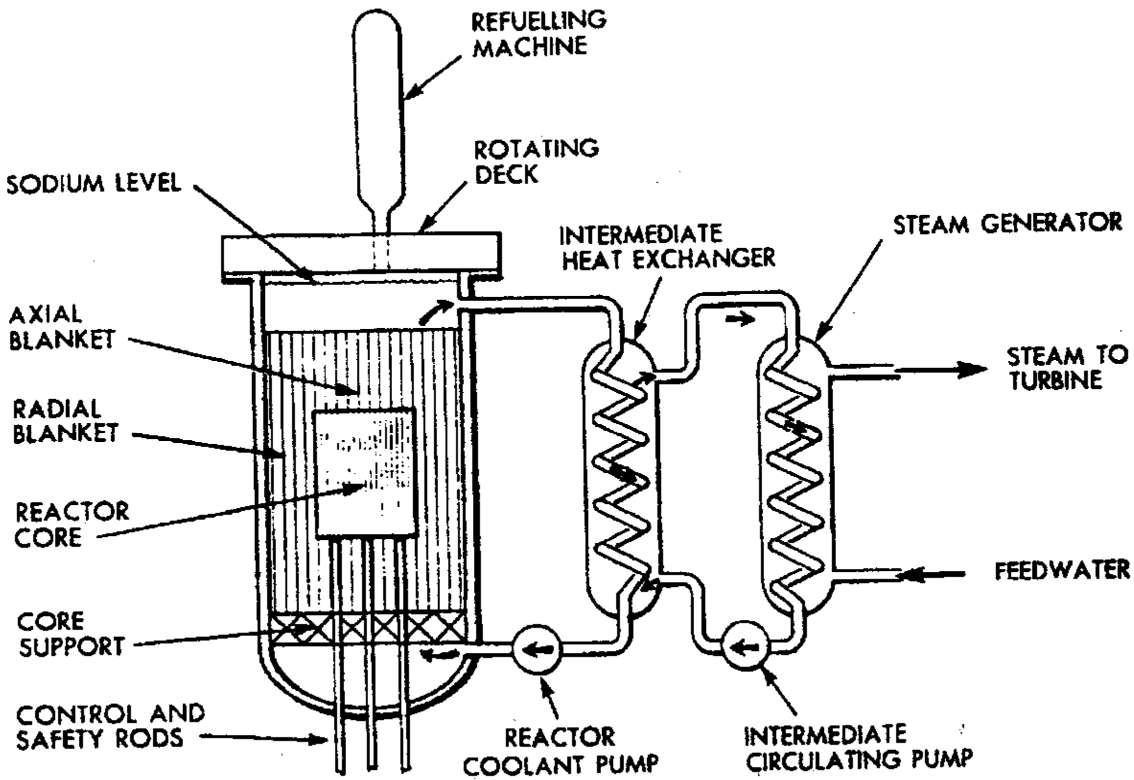

Figure 2.9 Schematic arrangement LMFBR

2.11.7 CANDU

This, as you know, is the generic name for the Canadian heavy water moderated, natural uranium type power reactor. You will notice that there is no coolant specified in this definition. This is because a variety of coolants can be used.

All CANDU reactors possess certain basic characteristics and features as follows:

Neutron Economy

This is the keystone of the concept. If natural uranium fuel is to be used economically, high burnups must be achieved, i. e., the megawatts extracted per kilogram of uranium must be high. This led to the choice of heavy water as the moderating medium since heavy water is by far the most neutron economic moderator available.

Pressure Tubes

While it is possible to use heavy water in a PWR type of pressure vessel reactor as both the coolant and moderator, the size of pressure vessel required is rather larger than for a PWR because the required volume of D20 moderator is much greater than the required volume of H20 moderator. The early studies of CANDU reactors were based on the pressure vessel approach and, in fact, NPD started out to be a pressure vessel type. It was, however, recognized that the size of pressure vessel manufacturable in Canada at that time would be quite limited, placing a definite limit on the power output achievable when the first commercial units were built. At the same time, the development of zirconium alloy (a neutron economical material) had proceeded to the point where it became possible to employ this material for pressure tubes. Before proceeding to describe the pressure tube approach, I should say that the pressure vessel approach was followed by Sweden and Germany for some years and is still being followed by Kraftwerk-Union for a plant they built in Argentina.

The pressure tube reactor concept can be described as follows. The reactor consists of an array of pressure tubes, generally arranged on a square lattice, which pass through, from end to end, a large cylindrical tank. The reactor fuel, in the form of cylindrical clusters of individual fuel rods, resides inside the pressure tubes. The coolant is pumped through the pressure tubes to cool the fuel. The fact that this coolant is generally at high pressure gives rise to the term "pressure tube".

The heavy water moderator is held in the large cylindrical tank which surrounds the pressure tubes. This large cylindrical tank is called the calandria. Because the coolant, and hence the pressure tubes, must operate at high temperature in a power reactor and because it is desirable to operate the moderator at low temperature to avoid the necessity of pressurizing the calandria, the pressure tubes must be insulated from the moderator. This is done by introducing a second tube which surrounds the pressure tube but is separated from it by a stagnant gas space. This second tube is called the calandria tube. This calandria tube is sealed at both ends to the calandria end plates or tubesheets, thereby completing the moderator containment.

With this arrangement the fuel coolant is completely separated from the moderator, permitting a free choice of coolants.

(iii) On-Power Fuelling

If natural uranium fuel is to be employed and high burnups achieved, neutrons must not be wasted needlessly. This is best achieved by introducing new fuel and removing old, burned-up fuel in a "continuous" manner since the excess reactivity possessed by the new fuel can be used to compensate for the loss of reactivity on the part of the old fuel, thereby extending its useful life.

The pressure tube approach lends itself to on-power refuelling since the fuel residing in individual pressure tubes can be changed without affecting other pressure tubes or the fuel in them.

(iv) Separate Moderator

As was mentioned earlier, the pressure tube approach used in CANDU reactors permits the heavy water moderator to be kept quite separate from the fuel coolant. This, in turn, permits the moderator to be operated at a low temperature, which has several advantages:

- the calandria can operate at atmospheric pressure, avoiding the need for a heavy, high pressure vessel
- the cold moderator can act as a valuable heat sink under certain accident conditions.
- since the moderator is cold it cannot add energy to the reactor containment under accident conditions. This reduces the total quantity of energy which the containment system must handle.

In the foregoing, I have described certain general features common to all CANDU reactors. I will now discuss the various types of CANDU reactors developed to date.

2.11.8 CANDU-PHW

This is the pressurized heavy water (PHW) cooled version. It was the first type developed and is by far the most widely used. While not inherently necessary, this version has to date always employed a horizontal reactor core orientation. Vertical versions have been studied a number of times but no clear incentive to switch to this orientation has been identified.

A schematic arrangement of the PHW version is shown in Figure 2.10. Pressurized (~ 1400 psi) heavy water coolant at~ 480°F is supplied to each fuel channel (an assembly consisting of the zirconium alloy pressure tube with an alloy steel fitting attached at each end) via an individual pipe, called a feeder pipe. As the coolant passes through the fuel channel it picks up heat from the fuel and leaves the channel at ~560°F. It is then conveyed to the outlet header via the outlet feeder pipe. From the outlet header, the coolant flows through the boiler heat exchangers where it is cooled back to ~ 480°F, its heat being given up to produce steam at ~ 600 psi which is fed to the turbine. The coolant then enters the circulating pumps which deliver it to the reactor inlet header and, thence, to the inlet feeder pipes.

A separate auxiliary circuit is employed to circulate the heavy water moderator through external heat exchangers. These reject to the station cooling water the heat generated in the moderator by the slowing down of the neutrons, by the effects of γ radiation, and also the heat leaking into the moderator across the insulation gaps between the calandria tubes and pressure tubes.

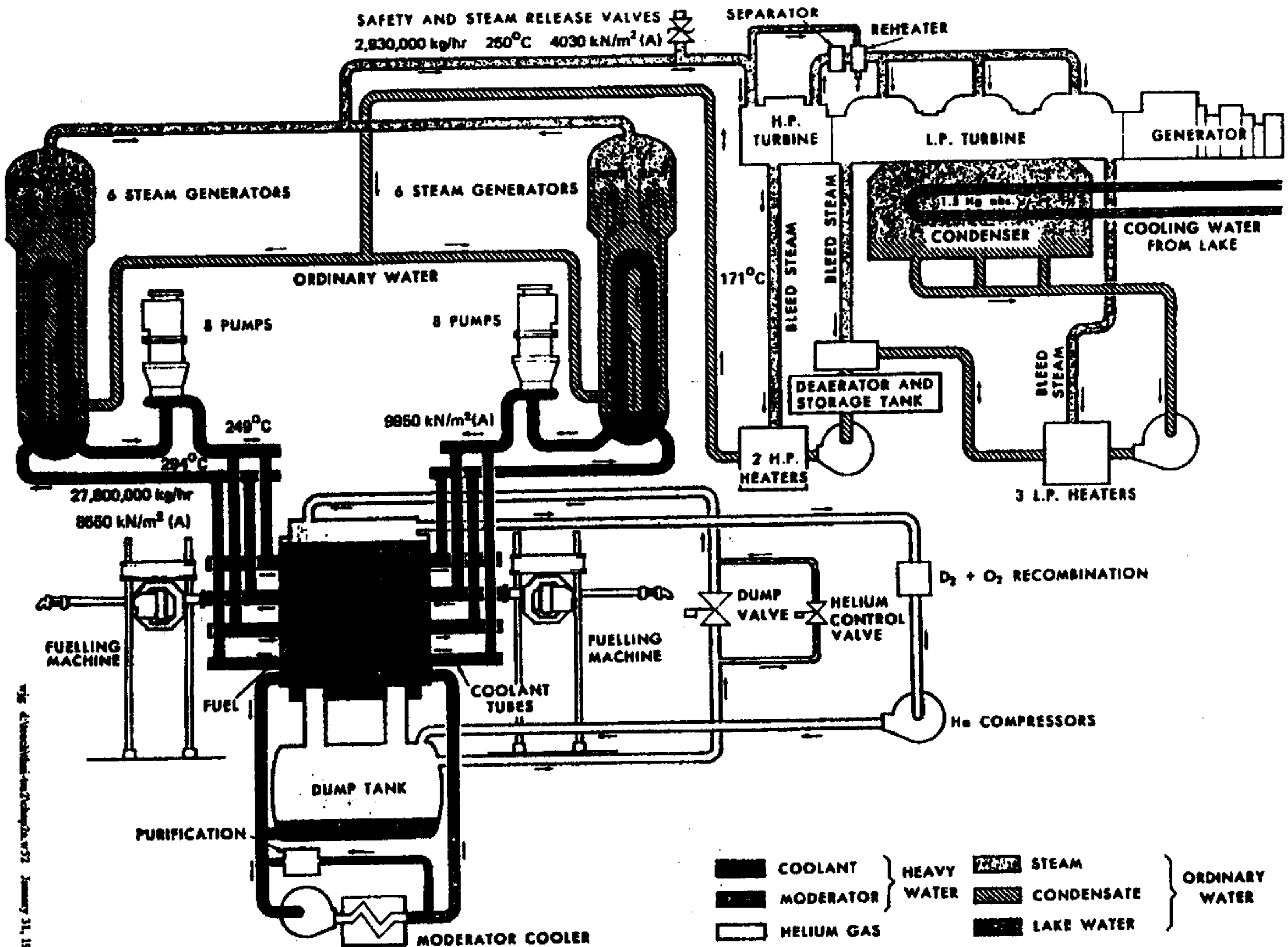

Figure 2.10 CANDU PWR schematic

2.11.9 CANDU-BLW

This was the second version of the basic CANDU concept to reach the prototype reactor stage (the 250 MWe Gentilly plant). Its major difference lies in the choice of coolant: boiling light (ordinary) water, hence BLW. Its reactor coolant and turbine systems are fundamentally the same as those of the BWR described earlier, i.e. , a direct cycle is employed.

For this version, a vertical orientation was chosen. There were a number of detailed considerations relating to the boiling coolant which led to this choice. It is likely that future CANDU-BLW reactors will retain this orientation. Design details of the reactor proper will be described later.

Figure 2.11 provides a schematic illustration of the concept. Ordinary water is pumped to the bottom of each fuel channel via an individual feeder pipe. As the water passes upwards and absorbs heat from the fuel, a fraction (~ 18%) is evaporated to steam. The resulting steam/ water mixture then flows to a conventional steam drum where the steam and water are separated. The steam then flows to the turbine and the water, mixed with incoming feedwater in the drum, flows down to the circulating pumps, completing the cycle.

The moderator cooling system is basically the same as for the PHW version.

The British have developed a similar version which they call the SGHWR (steam generating heavy water reactor). A 100 MWe prototype has been built and is in operation. It differs from our Gentilly prototype in that it uses slightly enriched fuel. This permits rather less heavy water moderator to be used, reducing capital costs. The fuelling costs are, however, somewhat higher. Another possible variation is one in which the enrichment is provided by plutonium which is produced as a by-product in the fuel used in our PHW reactors. This plutonium, as plutonium oxide, would be mixed with natural UO_2 in the fuel.

2.11.10 CANDU-OCR

A third version of the basic CANDU concept is one which would use an organic fluid as the coolant. It would be similar to the PHW concept except that the boilers would likely be of the "once-through" type with some steam superheating provided. This is made possible by the fact that the coolant temperature at the reactor outlet can be ~ 100°C higher than in the case of heavy water cooling.

The WR-1 experimental reactor at our Whiteshell establishment employed this concept except that the heat is "wasted", i. e. , no turbine was provided. The reactor operated with coolant conditions which are the same as would be employed in a commercial power plant.

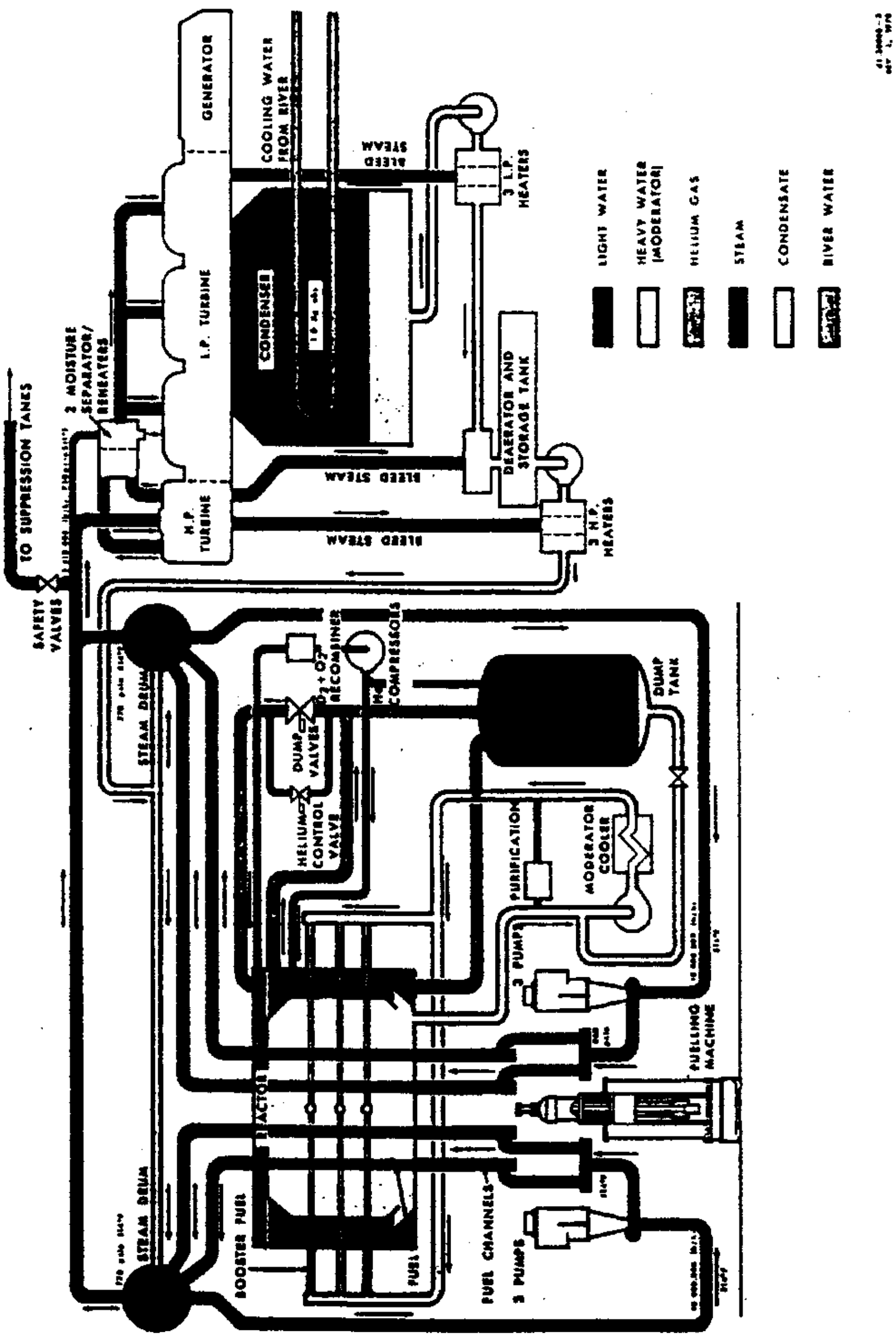

Figure 2.11 Simplified station flow diagram - CANDU BLW

2.11.11 A Comparison Between CANDU Reactors and Other Types

Currently the major competitors to the CANDU system are the light water reactors (PWRs and BWRs).

With regard to the competitive position between current commercial reactors, CANDU-PHW on one hand and the light water reactors on the other, neither type has a clear lead in all cases. This arises because of differences in basic characteristics.

From the standpoint of fuelling costs, the CANDU-PHW is the clear winner. This is, of course, because it can use natural uranium fuel whereas the light water reactors require enriched fuel. The enriched fuel is more expensive in several regards. Firstly, there is the cost of producing the enriched UO_2. This appears both as a "consumption" cost and as an added interest cost on the fuel while in manufacture, while resident in the reactor and while awaiting subsequent chemical reprocessing. Secondly, there is a manufacturing cost penalty because of the precautions necessary to avoid a criticality accident. Thirdly, there is a much more severe penalty should the fuel fail before achieving its full burnup life. Such failure is fundamentally more likely with enriched fuel because its "economic" life (burnup) needs to be approximately double that of natural fuel.

From the standpoint of capital costs, the picture is not as clear. The generally held view is that the capital cost of a light water reactor will be considerably lower. This is at least partly attributable to the difference in the way that heavy water and enriched fuel costs are accounted for in common utility practice. The former is treated as a normal plant depreciating capital asset whereas, in fact, it does not really depreciate. The latter is not considered as a capital asset. The fact is, of course, that a large amount of somebody's money is tied up in the enriched fuel. This is not, however, always utility money although the utility ultimately pays for it in terms of interest charges as pointed out earlier.

There is another significant difference which, while real, is not inherently the result of differences between the concepts. Relatively little advantage has been taken to date in the replication of design between CANDU plants. This is because relatively few have been built compared to PWRs. As a result, the CANDU reactors have been burdened with higher engineering costs and in costs arising from longer construction schedules because of relative inexperience. This difference is now diminished because we have built a strong technological base which will permit the replication of most design features from plant to plant.

There is only one way in which a utility can really answer the question as to which type is best for it and this is to go through a full comparative evaluation program based on its own requirements and financing position. Certainly, capital costs quoted in technical journals, newspapers, etc., are meaningless because they are, of necessity, quoted "out of context".

From a purely technical standpoint, one cannot say that one type of reactor has a clear advantage over the other, whether this be in terms of safety, or availability, or ease of operation, or what have you. For example, the use of heavy water at elevated temperatures and pressures for the coolant in the CANDU-PHW imposes strict requirements on coolant system leak-tightness and on systems for recovering leakage. Leakage is, however, not greatly more tolerable in the light water cooled reactors, primarily because of radioactive materials in the water. A, perhaps, compensating feature in another direction is that the on-load refuelling capability of the CANDU-PHW means that fuel defects are much more tolerable since the defective fuel can be readily removed. In the case of the light water reactors, the removal of defective fuel requires a plant shutdown of several weeks' duration.

One last point on the subject of comparisons. The CANDU-PHW is a "water reactor" as are the PWR's and BWR's. They therefore share many of the same advantages and problems. There is no question but that we, in developing the CANDU-PHW, have benefited greatly from much of the R & D work done for the light water reactors. Examples include UO_2 fuel, zirconium alloys for fuel cladding and reactor components, boiler heat exchangers, and main coolant pumps.