CHAPTER 1

Introduction to Nuclear Reactors

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***Summary**:*

This chapter provides a top-level introduction to nuclear reactors and surveys the world reactor situation. The various commercial large power producing reactors are identified and described against a brief background of nuclear reactor principles and key reactor components. The progressive expansion of nuclear power production is put into perspective in the global context. The operation of nuclear plants with respect to changing grid system demand is explained with some constraints identified. The chapter concludes with a brief review of safety and risk which are critical aspects in the design and operation of nuclear plants.

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1.1 Fission Energy

All atoms consist of a nucleus of protons and neutrons surrounded by a cloud of electrons. Generally, for elements of low atomic mass numbers, there are equal numbers of neutrons and protons in the nucleus because this arrangement represents the most stable configuration. As the atomic mass number increases, however, the increasing weak repulsive forces of similarly charged protons in the nucleus are compensated for by an increasing number of neutrons which contribute additional strong nuclear forces that bind adjacent protons and neutrons together. Therefore, at high atomic mass numbers, there are approximately one and one-half times as many neutrons as protons in the nucleus, as shown in Figure 1.

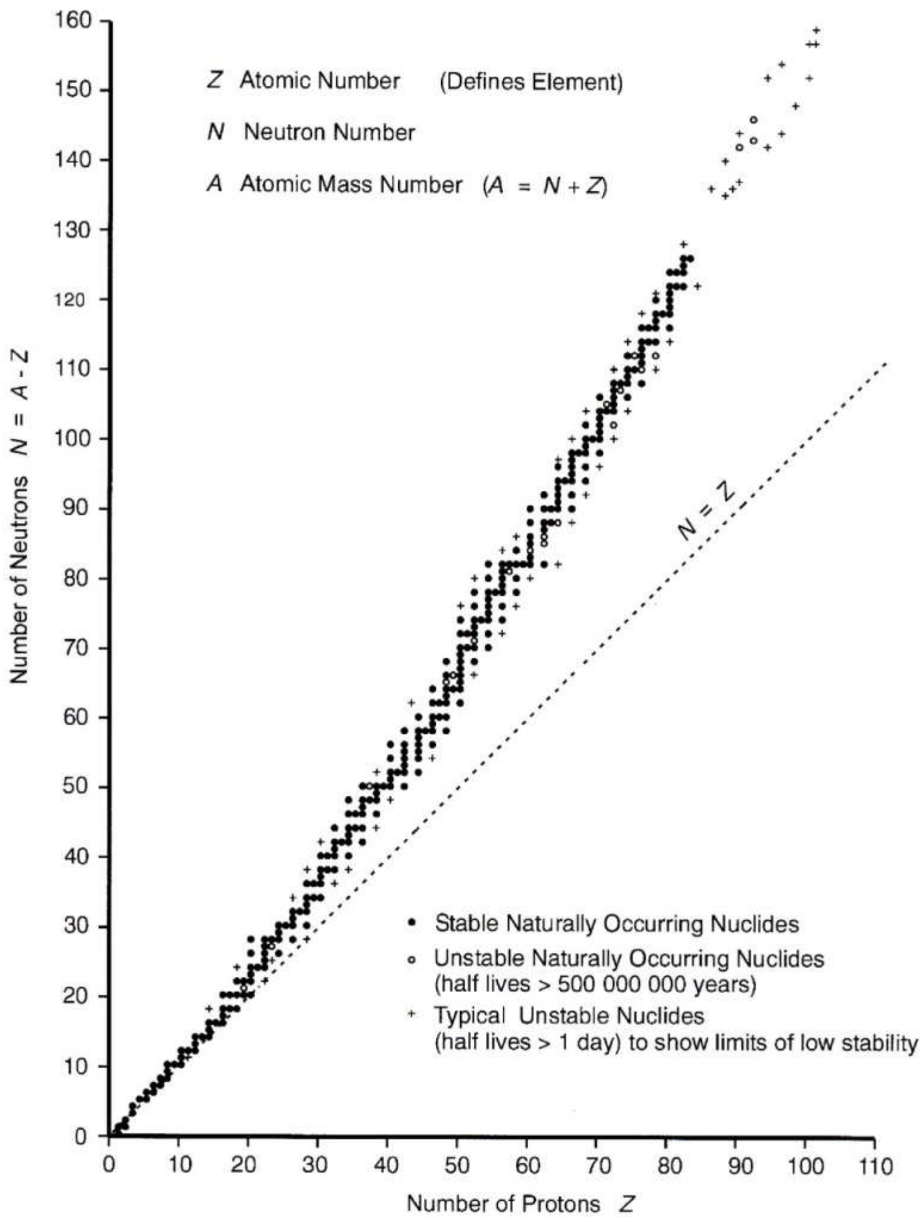

Figure 1 Range of stable and unstable nuclides

Should an element of high atomic mass number split, during a fission process, into two elements of lower atomic mass number, not as many neutrons will be required to create a stable configuration, and the surplus neutrons will be released.

The nucleons (neutrons and protons) in the nucleus are bound together by nuclear forces. When a nucleus is assembled, the binding energy attracting the individual nucleons is released. Conversely, energy is required to separate individual nucleons from one another or from the nucleus. This is analogous to the attraction of magnets to one another, but with a different type of force. This binding energy is a maximum per nucleon for elements near the middle of the range of atomic mass numbers and is somewhat less for elements with high atomic mass numbers, as shown in Figure 2. This means that, if a heavy element fissions or splits into two mid-range elements, the nucleons are bound together with a greater amount of binding energy per nucleon. The surplus energy is released as the nucleons come together more strongly. This is analogous to masses with potential energy falling into deeper holes and releasing this energy. Because of the well-known relationship between mass and energy, the release of binding energy is accompanied by a very slight decrease in the total mass of the constituents.

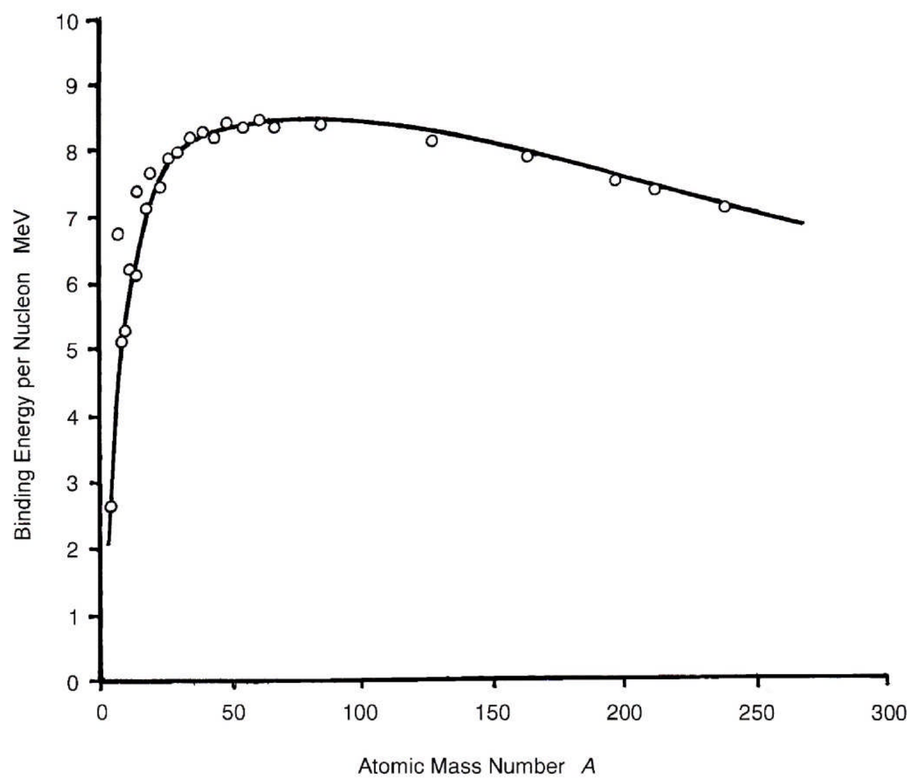

Figure 2 Binding energy per nucleon

Elements with high atomic mass numbers are unstable and become more so as the atomic mass number increases. If a neutron is added to the nucleus of a very heavy element, the additional binding energy brought into the nucleus may excite it to the extent that it fissions. This occurs in two isotopes of uranium and two isotopes of plutonium. Of these four fissile materials, only uranium-235 is naturally occurring. Furthermore natural uranium contains only 0.7 percent of U-235. Nevertheless, uranium-235 is the primary fuel of all commercial power producing nuclear reactors.

When a U-235 nucleus fissions, the release of binding energy amounts to about 200 MeV, or 32 x 10^{-12} J. Although a very small amount, this translates into a heat production rate of 950 MW if 1 kg of U-235 is totally consumed per day. In a typical nuclear power plant operating on a conventional steam cycle, this could be converted into an electrical power output of about 300 MW.

1.2 Nuclear Reactor Principles

Some heavy elements, such as uranium-235, can be induced to fission by adding a neutron to their nuclei. When fission occurs, the resultant lighter elements do not require as many neutrons in their nuclei to maintain a stable configuration and, on average, between two and three surplus neutrons are released. These neutrons can cause further fissioning of other U-235 nuclei and so establish a chain reaction. Such a reaction can be allowed to diverge, as in an atomic bomb, or be controlled, as in a nuclear reactor. Under steady state conditions, just one neutron on average from each fission should go on to produce another fission event.

When fission occurs, the release of energy drives the lighter elements or fission products and the surplus neutrons away from one another at high velocity. Most of the energy is thus transformed into kinetic energy carried by the fission products. As heavy strongly charged particles, they do not travel any significant distance and dissipate their kinetic energy in the fuel by interaction with other atoms, thus increasing the temperature of the fuel. The high energy fast neutrons, being uncharged, readily pass through the fuel and other reactor materials.

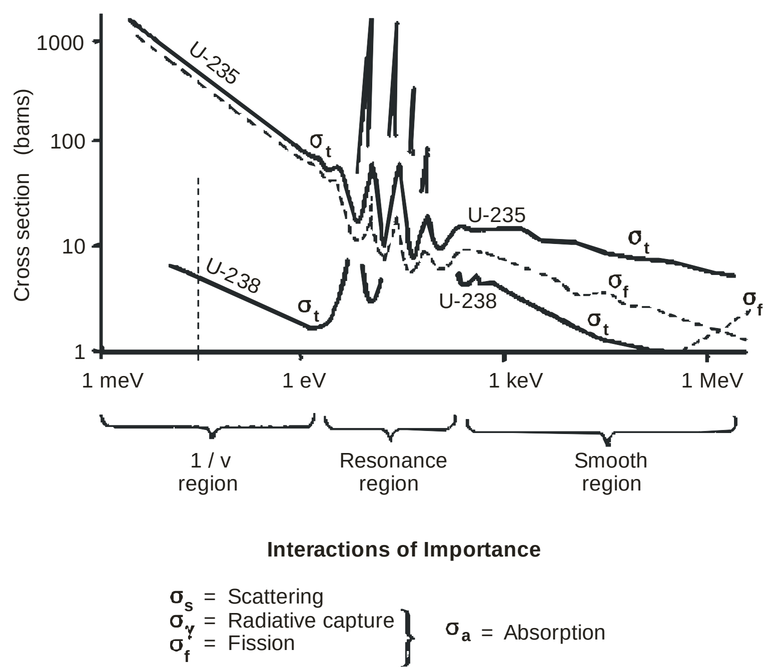

Figure 3 Fission and absorption characteristics of uranium

There are varying probabilities that they will be absorbed by different nuclei. The probability of absorption by another U-235 nucleus to cause fission increases if the neutron velocity is reduced, and therefore reducing neutron energies is advantageous. Figure 3 shows the variation of the fission cross section, that is, the probability of a fission reaction, of U-235 versus neutron energy as a dotted line. The probability increases by a factor of nearly 500 as the neutron energy is reduced from the fissioning release energy to the natural energy of neutrons in the local environment, a process known as thermalization. This can be achieved by allowing the neutrons to experience a series of non-absorbing collisions with light nuclei, which during an elastic collision receive some of the energy from the neutrons. The resulting low energy slow neutrons are then absorbed in U-235 nuclei to cause further fissions. Some elements that can slow down or moderate neutrons effectively without significant absorption are hydrogen, deuterium, helium, beryllium, and carbon. Hence, light water (H_2O), heavy water (D_2O), and graphite (C) all make good moderators in nuclear reactors.

The solid lines in Figure 3 show the total probability of absorption (resulting in both fission and capture) in U-235 and U-238. In the resonance region (where neutron and nucleus frequencies coincide) are large spikes with very high probability of capture for U-238, so it is best to avoid this region during the neutron slowing down process. This can be achieved to some degree by ensuring that the moderation process is carried out away from the fuel. This leads to a reactor consisting of a matrix of fuel elements within the moderator with a discrete distance between fuel elements. Furthermore, to promote heat transfer from the fuel, the fuel elements themselves consist of bundles of small fuel rods. This is illustrated diagrammatically in Figure 4.

Most of the heat from fission is generated by dissipation of the kinetic energy of the fission products. Because this occurs in the fuel near the point of fission, it follows that the fuel becomes the main source of heat in the reactor. To maintain a thermodynamic cycle to produce work, this heat must be removed continuously as it is produced. A suitable coolant is therefore required to flow over the fuel elements and remove the heat. The coolant must not readily absorb neutrons and must have suitable thermal properties. Coolants such as light water, heavy water, helium, and carbon dioxide meet these requirements.

Finally, to ensure steady state operation, the number of neutrons allowed to go on to produce fission must be the same as the number in preceding generations of neutrons. To achieve the required balance, the reactor as a whole is designed to generate excess neutrons in each generation and to have a system for absorbing the excess so as to maintain and control the reactor at a steady load. This also enables the number of neutrons in successive generations to be increased when more power is required or to be decreased when power must be reduced. Such control is usually achieved by having movable neutron-absorbing control rods partially inserted into the reactor. By fully inserting the control rods, the reactor can be shut down.

Therefore, a typical nuclear reactor consists of the following main components as shown in Figure 4:

- Fuel in which fission occurs and heat is generated
- Moderator to reduce the energy of the neutrons
- Coolant to remove the heat from the fuel elements
- Control rods to maintain the proper neutron balance.

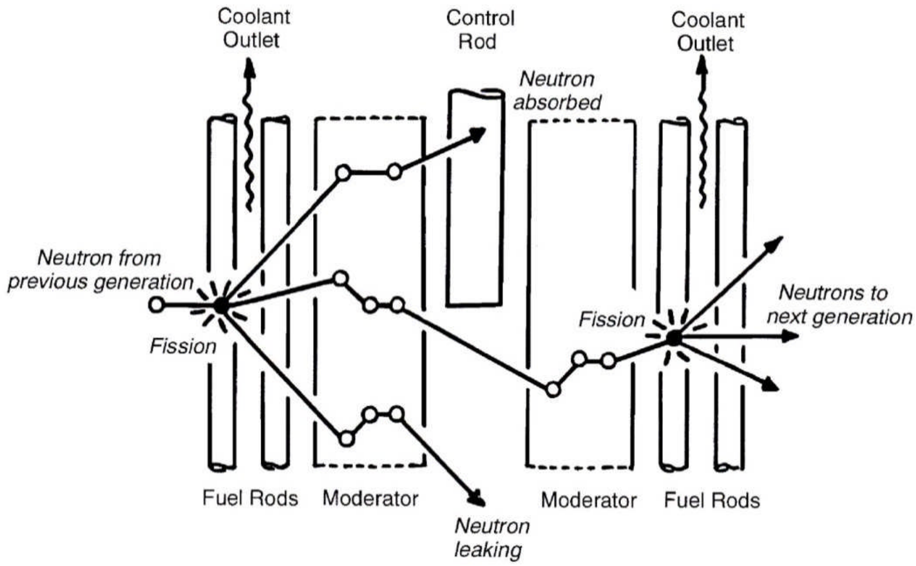

Figure 4 Nuclear reactor components

These components are usually arranged in a two-dimensional matrix so that neutrons generated by fission in one fuel rod pass through the moderator before entering the next fuel rod. The spacing of the fuel rods depends upon the moderator properties and the distance required to reduce the energy of the neutrons. The fuel rods are made small enough to promote heat removal and are surrounded by coolant. Often the coolant serves as a moderator as well. The control rods are made to penetrate between the fuel rods to capture excess neutrons effectively. Naturally, some neutrons leak through the boundaries of the system, and many are absorbed by the reactor materials or in the fuel without causing fission, meaning that the control rods do not have to absorb a significantly large number of neutrons compared with the number causing fission. The actual configuration of the matrix and the spacing of the fuel rods depend upon the fuel and moderator characteristics.

2 Reactor Types

2.1 Prototype Reactors

As nuclear reactors developed around the key elements of fuel, moderator, and coolant, many different types were proposed and constructed as demonstration models in an endeavour to prove their technical and commercial viability. Some designs had serious technical problems, while others had uniquely advanced features. However, due to arbitrary political decisions, one or two good designs did not go beyond the prototype stage, while others with insurmountable technical difficulties were abandoned. Nevertheless, some prototypes did operate successfully for many years, providing valuable technical and operational experience while producing power. Ultimately, the field narrowed to certain proven designs which were adopted on a commercial basis. Currently, there are six main types, which are listed below in order of numbers in service. The first five have proven to be commercially viable, while the sixth can be considered to be still in the prototype stage, but to hold promise for the future as a reactor which can breed new fissile fuel:

PWR	Pressurized Water Reactor
BWR	Boiling Water Reactor
PHWR	Pressurized Heavy Water Reactor (CANDU)
GCR	Gas Cooled Reactor
LGR	Liquid Graphite Reactor
LMFBR	Liquid Metal Fast Breeder Reactor.

2.2 Commercial Reactors

The reactors listed above will be described briefly in the following paragraphs for comparison, starting with the simplest concept.

2.2.1 Boiling water reactor (BWR)

The boiling water reactor consists of a pressure vessel containing the fuel rods in vertical elements, with each element surrounded by a channel. The channels are flooded with light water which serves as the moderator and coolant. Pumps circulate water upwards through the channels, and steam is generated within the channels. The exit steam quality is about 13%, and the steam is separated from the water in cyclone separators above the reactor core. This saturated steam is sent directly to the steam turbines.

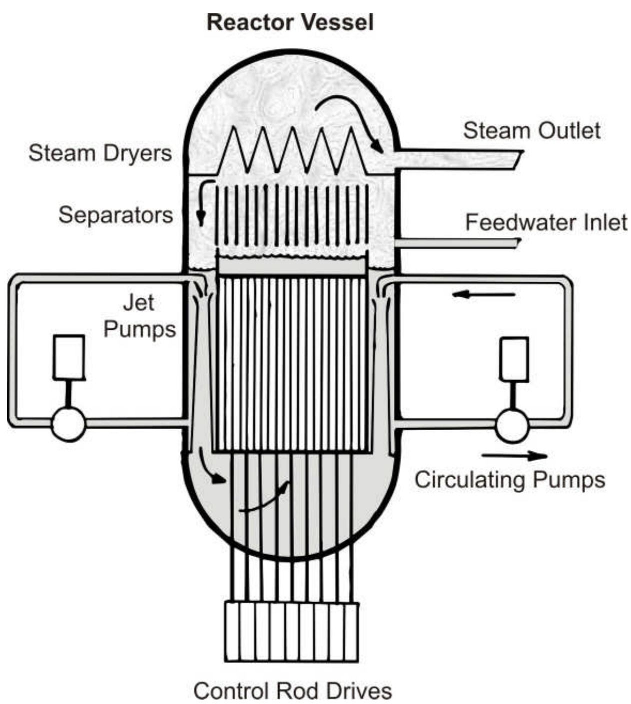

Figure 5 Diagrammatic cross section of a typical BWR

This simple direct cycle has the disadvantage that steam from the reactor is passed through the turbine and the condensate is returned through the feedwater heaters, making the whole steam circuit slightly radioactive during operation. A simplified diagram of a boiling water reactor is given in Figure 5.

2.2.2 Pressurized water reactor (PWR)

The pressurized water reactor overcomes the problem of a slightly radioactive steam circuit by having an intermediate heat exchanger to separate the reactor coolant circuit from the turbine steam circuit. Steam is generated in this steam generator and is sent to the turbine as saturated steam under conditions similar to those in the boiling water reactor. The reactor coolant circuit is maintained at high pressure to prevent any boiling in the reactor and operates at a slightly higher temperature than the BWR to promote heat transfer to the secondary steam circuit. Because no boiling occurs in the reactor core, it is more compact and does not require channels. The fuel rods of the individual elements form a continuous vertical matrix in the core. This is flooded with an upward flow of circulating light water which serves as coolant and moderator. A simplified diagram of a pressurized water reactor is given in Figure 6.

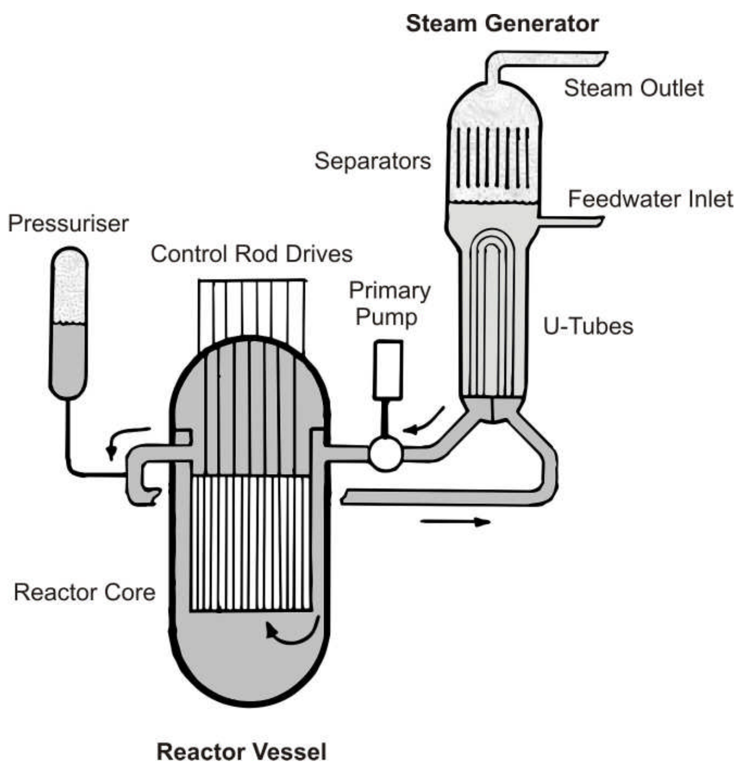

Figure 6 Diagrammatic cross section of a typical PWR

2.2.3 Pressurized heavy water reactor (PHWR) (CANDU)

The pressurized heavy water reactor is similar to the pressurized water reactor in that it has a primary coolant circuit and secondary steam circuit with a steam generator producing saturated steam. The reactor, however, is different in that it has individual pressure tubes passing horizontally through the reactor core. Fuel rods in the form of bundles are contained within these pressure tubes. The heavy water coolant flows through these tubes and is then circulated through the steam generator. The pressure tubes are surrounded on the outside by heavy water at low pressure and low temperature, which serves as the moderator. This is contained in a large vessel or calandria at ambient pressure. The advantage of this system is that it does not require a very heavy pressure vessel as in the BWR and PWR. The Canadian version is known as the CANDU. Other variations such as the steam generating heavy water reactor (SGHWR) had light water coolant and vertical channels and generated steam in the core. A simplified diagram of a pressurized heavy water reactor is given in Figure 7.

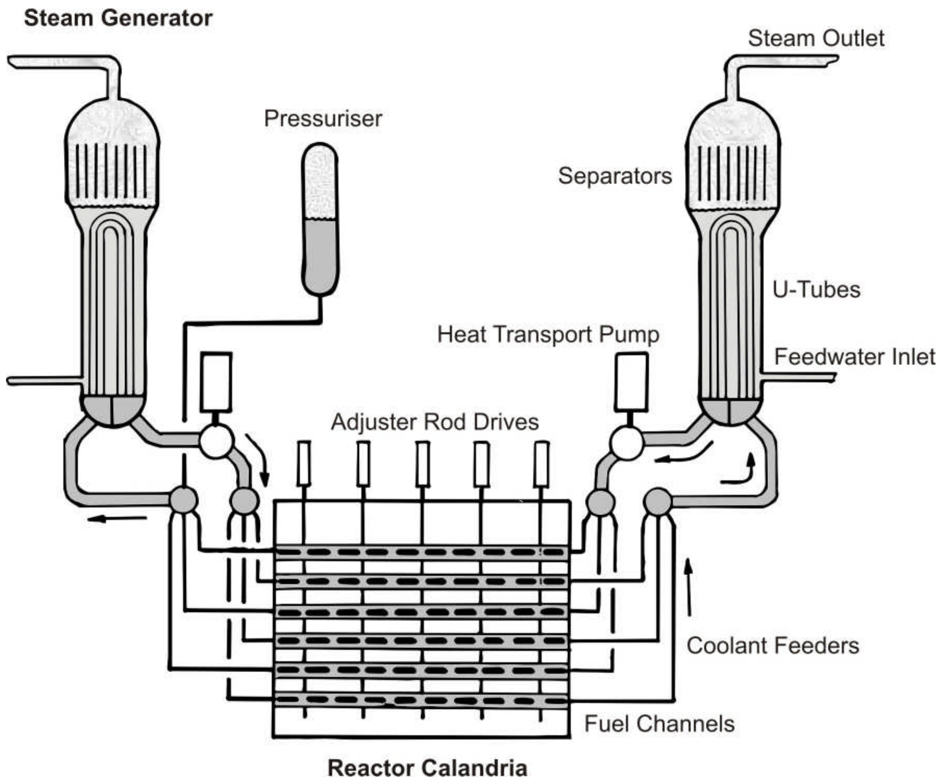

Figure 7 Diagrammatic cross section of a typical PHWR (CANDU)

2.2.4 Gas cooled reactor (GCR)

The gas cooled reactor, like the pressurized water reactor, has separate reactor coolant and steam generating circuits with an intermediate steam generator. The coolant, however, is gas, which enables higher coolant temperatures to be achieved. This enables superheated steam to be generated and a high efficiency steam circuit similar to that of a fossil fuel fired plant to be used. Due to the high temperatures and large core volume required with a gas coolant, graphite is more suitable than water as a moderator. This is installed in the form of blocks with holes to form channels into which the fuel elements are placed and through which the gas coolant flows. The early reactors of this type were known as Magnox from the type of fuel used, and later ones were called advanced gas cooled reactors (AGRs), which run at higher temperatures with a different type of fuel. By changing the coolant from carbon dioxide to helium, even higher temperatures could be achieved, hence the designation *high temperature reactor* (HTR). This type gave the ultimate advantage of large gas cooled reactors, which have a cycle efficiency as high as any fossil fuel fired plant.

Later developments in this field, such as the pebble bed modular reactor (PBMR), use a direct cycle, with the helium coolant passing directly to a helium turbine with the promise of even better cycle efficiency. A simplified diagram of an advanced gas cooled reactor is given in Figure 8.

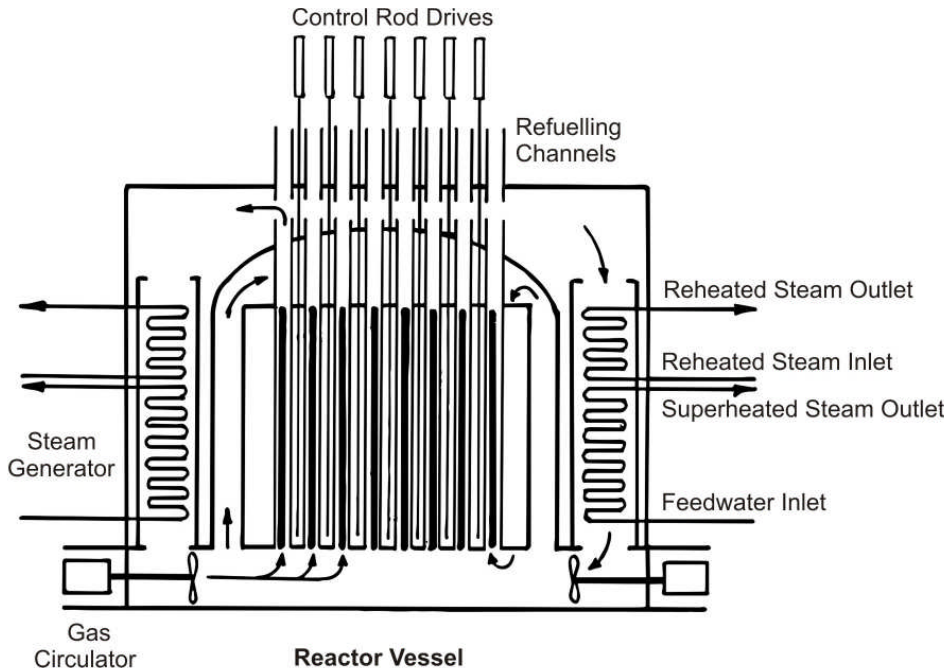

Figure 8 Diagrammatic cross section of a typical AGR

2.2.5 Liquid graphite reactor (LGR)

The liquid graphite reactor has a graphite block core similar to that of a gas cooled reactor. Pressure tubes similar to those of the pressurized heavy water reactor pass through vertical holes in the graphite, which serves as the moderator. Fuel elements made up of clusters of rods are located within these vertical tubes, and light water coolant flows upwards over the fuel and within the tubes. Boiling occurs in the fuel as in the boiling water reactor. The resulting steam is separated from the water in an external drum and the water recirculated. As with the boiling water reactor, saturated steam is sent to the turbine. This type of reactor of Russian design is known as the RBMK. A simplified diagram of a liquid graphite reactor is given in Figure 9.

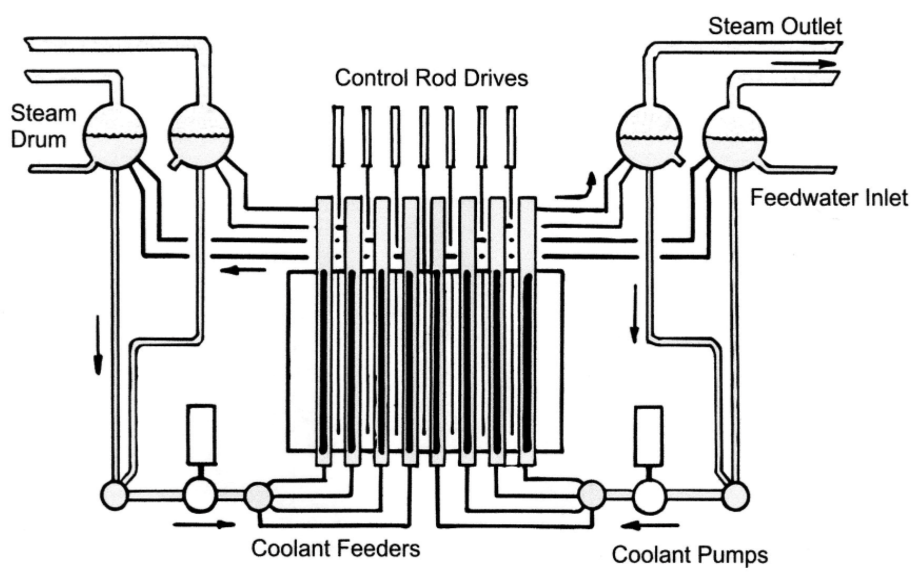

Figure 9 Diagrammatic cross section of a typical RBMK (LGR)

2.2.6 Liquid metal fast breeder reactor (LMFBR)

The liquid metal fast breeder reactor is different from the others in many ways. The coolant is a liquid metal, usually sodium, which flows directly over the fuel rods of the vertical fuel elements in an upward direction. The heat carried in this primary circuit is transferred to a secondary sodium circuit in a heat exchanger. This secondary circuit then transfers the heat to the steam circuit in a steam generator. Because sodium remains a liquid at high temperatures without being pressurized, heavy pressure vessels are not required, and high temperature superheated steam can be produced to provide good cycle efficiency. There is no moderator to reduce the energy of the neutrons, hence the term “fast”, and surplus neutrons are used to convert non-fissile material into fissile fuel, hence the term “breeder”. This type of reactor has not yet reached the commercially viable stage, and therefore there is no typical design.

3 World-Wide Commercial Power Reactors

3.1 Reactor Development

Following the Second World War, the development of nuclear reactors followed different paths in different countries depending upon the facilities developed during the war and the perceived military needs of certain countries following the war. The first nuclear reactors served to generate plutonium-239, another fissile material formed when uranium-238, the major constituent of natural uranium, absorbs excess neutrons. Pu-239 could be easily separated from the original uranium fuel and was needed for atomic bombs. U-235 is also used in nuclear weapons, but is difficult to separate from U-238.

The United States had isotope separation facilities for uranium and therefore was able to pursue the development of light water reactors requiring the use of uranium fuel slightly enriched in U-235. The United Kingdom did not have such facilities and therefore was forced to develop reactors using natural uranium. Canada had developed expertise in heavy water technology as a result of the war effort and was therefore well positioned to develop heavy water moderated reactors also using natural uranium.

When using natural uranium as a fuel, the low concentration of U-235 and the absorption of neutrons in U-238 necessitate use of a moderator with an extremely low absorption cross section to establish a continuous chain reaction. Only heavy water (deuterium and oxygen) and graphite (carbon) have the required physical and neutron properties. This led Britain to develop graphite moderated natural uranium metal fueled reactors with carbon dioxide as a coolant. Later, when they had uranium enrichment facilities, they were able to switch to uranium dioxide as the fuel. Canada, which had supplies of heavy water, was able to develop heavy water moderated natural uranium dioxide fueled reactors. Heavy water is the only moderator that can be used directly with natural uranium dioxide fuel to sustain the required fission chain reaction. Although heavy water is an excellent moderator, especially with respect to neutron absorption, it is very expensive to separate from ordinary water. If enriched uranium is available, as was the case in the United States, there is wider scope in the choice of a moderator because more neutron absorption can be tolerated. Therefore, light water (hydrogen and oxygen) can be used as a moderator. An advantage of light water as a moderator is that it is very effective per unit distance in slowing down neutrons, leading to a smaller moderator volume and a more compact reactor than with the other moderators. However, light water has the inherent disadvantage of absorbing a greater fraction of neutrons than graphite or heavy water.

Of the three moderators mentioned above, graphite is the least effective in reducing neutron energy and requires the largest volume. Graphite moderated reactors are therefore the largest, leading to high capital costs. Considering capital cost, moderator costs, and enriched fuel supply and cost, all three of these became economically viable in their respective countries, and commercial reactors for power plants subsequently evolved.

Over time, enriched uranium became more easily available, and commercial reactors were sold to other countries which did not have fuel enrichment resources, leading to a free choice of reactors. It is of interest to review the world use of nuclear reactors for electric power production.

3.2 Commercial Reactors in Service

Table 1 shows the number, type, and output of reactors operating and committed or under construction in different countries. The first column under the two latter headings is for reactors currently in service and the second column for those reactors to be added in the coming years. This table is representative of current commercial technology because most prototype and early commercial reactors have served their useful life and been decommissioned.

Table 1 Nuclear reactors in service and planned*

BWR–Boiling (Light) Water Reactor

GCR–Gas Cooled Reactor

LGR–Light Water Graphite Reactor

LMFBR–Liquid Metal Fast Breeder Reactor

PWR–Pressurized (Light) Water Reactor

PHWR–Pressurized Heavy Water Reactor

Country	Reactor Type	Number Operational + Planned		Existing Capacity + New Capacity (MWe)	
Argentina	PHWR	2	1	935	692
Armenia	PWR	1		375	
Belgium	PWR	7		5 885	
Brazil	PWR	2	1	1 901	1 275
Bulgaria	PWR	2	2	1 906	2 000
Canada	PHWR	22		15 137	
China	PWR	12	41	9 748	42 230
	PHWR	2		1 300	
	GCR		1		200
	LMFBR		1		20
Czech Republic	PWR	6		3 678	
Finland	PWR	2	1	976	1 600
	BWR	2		1 740	
France	PWR	58	1	63 130	1 600
Germany	PWR	7		9 486	
	BWR	2		2 572	
Hungary	PWR	4		1 889	
India	PWR		2		1 834
	BWR	2		300	
	PHWR	18	4	4 091	2 560
	LMFBR		1		500
Iran	PWR	1		915	
Japan	PWR	24		19 286	
	BWR	26	2	24 818	2 756
Kazakhstan	LMFBR	1		70	

Country	Reactor Type	Number Operational + Planned		Existing Capacity + New Capacity (MWe)	
Mexico	BWR	2		1 300	
Netherlands	PWR	1		487	
Pakistan	PWR	2	1	600	300
	PHWR	1		125	
Romania	PHWR	2	3	1 300	1 869
Russia	PWR	16	11	11 914	9 810
	LGR	15		10 219	
	LMFBR	1	1	560	750
Slovakia	PWR	4	2	1 816	810
Slovenia	PWR	1		666	
South Africa	PWR	2		1 800	
South Korea	PWR	17	7	15 975	8 600
	PHWR	4		2 722	
Spain	PWR	6		6 004	
	BWR	2		1 510	
Sweden	PWR	3		2 799	
	BWR	7		6 504	
Switzerland	PWR	3		1 700	
	BWR	2		1 538	
Taiwan	PWR	2		1 780	
	BWR	4	2	3 104	2 600
Turkey	PWR		4		4 600
Ukraine	PWR	15	3	13 107	2 850
United Arab Emirates	PWR		4		5 600
United Kingdom	PWR	1		1 188	
	GCR	17		8 732	
United States	PWR	69	8	68 459	8 990
	BWR	35	2	34 935	2 700

* Adapted from Nuclear News, March 2012

It is evident from Table 1 that certain types of reactors became dominant in certain countries, particularly in those developing their own reactors, for example, PWRs and BWRs in the United States, GCRs in the United Kingdom, PHWRs in Canada, and LGRs in Russia. However, in later years, certain types of reactors have been favoured by countries without their own developmental program, leading to the spread of some of these types to other countries. The PWR, however, has become the most common type and is currently produced by various manufacturers around the world.

3.3 Recent Past and Near Future

Table 2 gives a summary of the situation at the beginning of 2012. It shows that PWRs make up 61% of operational reactors and 67% of total nuclear capacity, due mainly to their larger size relative to PHWRs and GCRs.

Table 2 Nuclear reactors in service in 2012*

Reactor Type	Number in Service	%	Total Capacity (MWe)	%
PWR	267	61	246 555	67
BWR	84	19	78 321	21
PHWR	51	12	25 610	7
GCR	17	4	8 732	2
LGR	15	4	10 219	3
LMFBR	1	0	560	0
TOTAL	435	100	369 997	100

*Adapted from Nuclear News, March 2012.

A similar review done in 1998 indicated that during the intervening 14 years, the number of reactors in service had only increased by two, but that total capacity had increased by 11%. This was due primarily to the decommissioning of smaller older reactors which had reached the end of their economic life. It does not reflect a lack of new plants coming on-line because the number of PWRs increased by 17, even though 10 were taken out of service during this period.

Looking forward over 10 years to 2022, Table 3 shows the total number of reactors currently in service and committed for commissioning or construction. Although the table does not consider possible decommissioning of some reactors, it also does not take account of new orders which will be completed before 2022. In this context, it should be noted that 80% of new construction is due to be on-line by 2017, leaving time for new commitments which will likely overcompensate for possible decommissionings.

Table 3 shows a dramatic increase of 25% and 29% respectively in the number of reactors in service and available capacity over the following 10 years.

Table 3 Projected reactors in service in 2022*

Reactor Type	Number in Service	Change (%)	Total Capacity (MWe)	Change (%)
PWR	356	+33	339 569	+38
BWR	90	+7	86 377	+10
PHWR	59	+16	30 722	+20
GCR	18	+6	8 932	+2
LGR	15	0	10 219	0
LMFBR	5	+400	2 076	+271
TOTAL	543	+25	477 895	+29

*Adapted from Nuclear News, March 2012

3.4 Status of Large Reactors

Table 4 New large reactors (1000 MWe and greater)*

Country	Reactor Type	Number Committed	New Capacity (MWe)
Brazil	PWR	1	1 275
Bulgaria	PWR	2	2 000
China	PWR	38	36 400
Finland	PWR	1	1 600
France	PWR	1	1 600
Russia	PWR	6	6 900
South Korea	PWR	7	8 600
Taiwan	PWR	2	2 600
Turkey	PWR	4	4 600
United Arab Emirates	PWR	4	5 600
United States	PWR	8	8 990
	BWR	2	2 700

*Data extracted from Nuclear News, March 2012

Most reactors currently under construction or on order are of large capacity, that is, 1000 MWe or greater. Considering only reactors of this size, Table 4 shows the type of reactor, total capacity, and country where these are being built or to be built.

For comparison, the number of large units taken out of service to date is given in Table 5. This table shows the capacity, date taken out of service, and number of years in service.

Table 5 Decommissioned large reactors (1000 MWe and greater)*

Country	Reactor Type	Capacity (MWe)	Date	Years of Service
France	LMFBR	1 200	1998	12
Germany	PWR	1 219	2001	14
	BWR	1 346	2011	27
	PWR	1 345	2011	32
	PWR	1 240	2011	34
	PWR	1 167	2011	36
United States	PWR	1,095	1992	16
	PWR	1 040	1998	24
	PWR	1 040	1998	25
TOTAL		10 692		220

*Data extracted from Nuclear News, March 2012

The average service life of these excluding the LMFBR is 26 years. Typically with refurbishment the life expectancy of such reactors would be at least double this.

4 Power Production

4.1 Energy Transfer

Heat, energy, and work all have the same units (joules), but somewhat different meanings, and power is the rate of doing work (watts). What is important is that not all heat can be converted into work. Although the first law of thermodynamics states that all work can be dissipated as heat, the second law states that not all heat can be converted into work. In a typical water cooled nuclear plant (CANDU, PWR and BWR) approximately 30% of the heat released by the fuel is ultimately converted into electrical energy. The rest must be discharged as low grade (low temperature) heat.

During the fission process, heat is generated in the fuel. This heat is removed by the reactor coolant flowing over the fuel rods and transported in the primary circuit to the steam generator, where it flows inside the tubes and its heat is transferred to the secondary circuit through the walls of the tubes in the steam generator. Water in the secondary circuit outside the tubes absorbs this heat and is converted into steam under saturated conditions. This steam is passed to the turbine where it expands to low pressure while being directed onto the turbine blades and in so doing transfers its energy to the turbine rotor. The rotor drives the electrical generator which produces electric power. The exhaust steam is condensed by cooling water passing through the condenser tubes and in so doing discharges the bulk of the heat which cannot be converted into work. The condensate is returned to the steam generator after being preheated in the feedwater heating system. A simple flow diagram for a nuclear plant is shown in Figure 10. Note that in the BWR and RBMK, steam is generated directly in the reactor, so there is no separate steam generator.

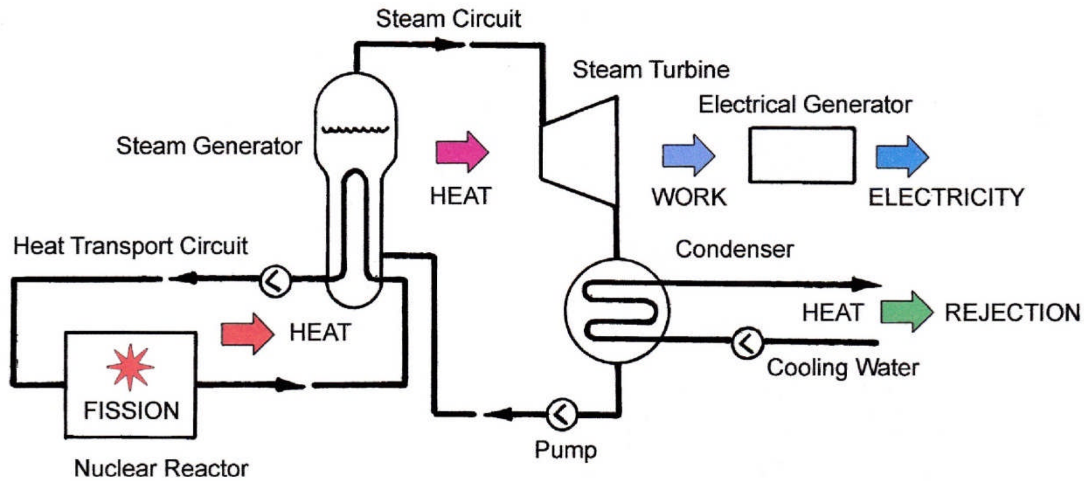

Figure 10 Simplified flow diagram of a nuclear plant

4.2 Power Output

Consider a very simple system with a nuclear reactor, steam generator, and turbine generator supplying electric power to an isolated electrical grid, as shown in Figure 11. This power must be generated at the moment it is required by the consumers connected to the grid. Power production must follow demand exactly, and any mismatch will cause the grid frequency to fall or rise as demand increases or decreases. A basic control system works as follows to maintain appropriate power output from the plant. In the event of an increase in demand the mismatch will cause the grid frequency to fall. Because the turbine generator is synchronized to the grid, its speed will drop accordingly. This will be sensed by the turbine governor which will open the governor valves to admit more steam to increase the power output of the turbine generator. The additional flow of steam to the turbine will cause a reduction in steam pressure in the steam generator. This in turn will be sensed by the reactor regulating system which will withdraw control rods from the reactor core until the increased fission rate generates sufficient additional heat to restore the steam generator pressure. In the event of a decrease in demand the reverse occurs. This is known as the reactor following or turbine leading mode of operation (or normal mode in some plants because it is a natural way of maintaining stable conditions).

Such a system, however, cannot maintain a specified frequency (60 Hz in North America) exactly without large unstable oscillations, and therefore a certain speed droop is incorporated into the turbine governor. This enables a progressive increase in governor valve opening (steam flow) as the turbine speed (grid frequency) falls. A typical droop setting is 4%, which means that, if the turbine was initially at zero load and full speed, its speed would have to drop to 96% before the governor valve would be fully open. Such a speed is not acceptable for the turbine due to possible blade vibration, nor to the grid due to loss in speed of connected motors. Therefore, the governor is adjusted to bring the speed back to 100% at full load. In the event of a turbine trip or load reduction to zero under these conditions, the reverse would occur, and the turbine speed would rise to 104% of full speed.

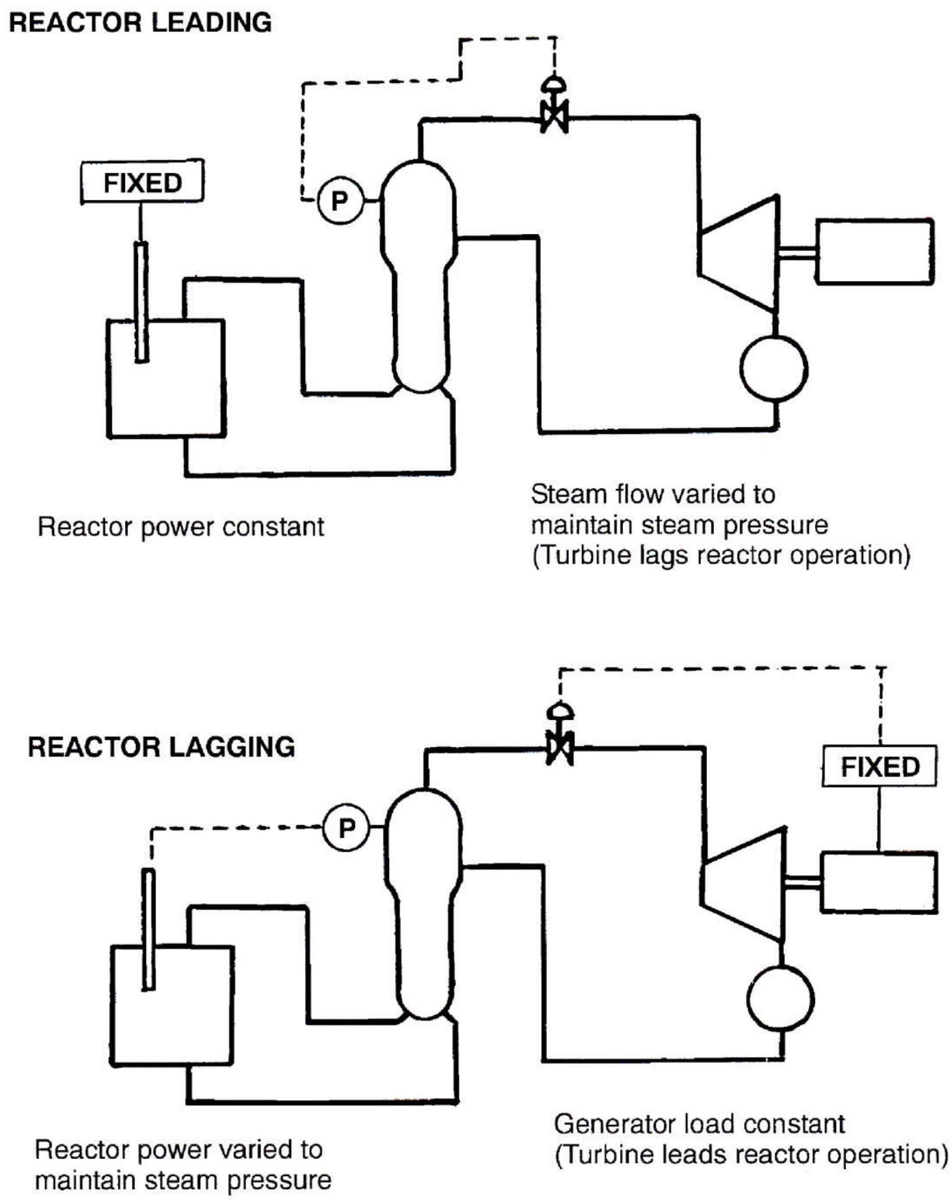

Figure 11 Lagging and leading plant operation

In the mode of operation just described, the nuclear reactor output follows the electrical grid demand, and therefore its power level oscillates continuously. This can have adverse effects depending upon the type and design of the reactor. Excessive oscillations impose temperature transients on the fuel, which can cause premature failures of the fuel cladding. Large oscillations near full power can cause power limits to be exceeded, thus tripping the reactor and losing power production as well as imposing restart transients on it and the turbine. Due to high capital cost and low fuel cost it is desirable to run nuclear reactors at full power most of the time. An alternative mode of operation is therefore often used at nuclear plants where the reactor power output is fixed. This is known as reactor leading or turbine lagging operation. To maintain stable operation, the reactor power is controlled at a given value by measuring the neutron flux and adjusting the control rods accordingly. Pressure in the steam generator must be maintained at the proper value to ensure stable conditions in the reactor coolant circuit. This is done by opening or closing the turbine governor valves to control steam flow from the generators. The turbine then delivers power according to the steam flow and the generator sends this power into the grid system regardless of the grid frequency. The grid frequency must then be controlled by other turbine generators which feed into the grid system and operate in the turbine leading mode.

By referring to the figure showing the two modes of operation, it can be seen that steam generator pressure is a key control parameter in both modes. This highlights the importance of the steam generator, where a balance of heat input and heat output must be maintained to maintain pressure. Furthermore, the difference in temperature between the primary coolant and the secondary working fluid determines the rate of heat transfer. Hence the reactor coolant temperature is determined by saturation temperature and thus by steam generator pressure.

4.3 Operating Constraints

During operation, parts of the reactor, steam system, and turbine are subject to high temperatures. If these parts have thick walls or have a substantial solid mass they will likely suffer thermal stress during the heating and cooling that arises during startup and shutdown and also during load transients. If a thick component is heated on one side, that side will tend to expand. If the side is constrained by the still cold base material so that it cannot expand an internal stress will be set up. The reverse happens during cooling. Therefore, large rigid components which are subject to transient and uneven heating and cooling will suffer low cycle fatigue damage and may ultimately fail. This effect can be minimized by slow heating and cooling to reduce temperature differences in single components such as reactor pressure vessels, steam generators, steam pipes, turbine casings and turbine rotors. This means that all these components must be preheated slowly before startup and the unit must be loaded slowly. Similarly, load changes on the unit should be carried out slowly. This requirement imposes operating restrictions on reactors and turbines. Generally, the larger the unit, the longer the time to start it up and load it. This makes large units less flexible in operation than smaller units.

In the reactor, temperature transients in the fuel cause structural changes within the fuel and stress on the cladding and therefore must be minimized to avoid premature fuel failures. In the turbine, uneven heating and cooling of the rotor can cause bending, which in turn causes excessive vibration and in the extreme case contact with the casing and therefore must be monitored very carefully.

A nuclear reactor is also subject to xenon transients which may inhibit operation for a certain period. When the load on a reactor is suddenly reduced, xenon, a fission product that absorbs neutrons very strongly, builds up in the fuel and may force a reactor shutdown. The xenon eventually decays after about 40 hours, and the reactor can be restarted. During this time, there is a loss in electrical power production, and the larger the unit, the more serious is this loss in revenue generating output.

4.4 Fuel Burnup

As the fuel in the reactor is used up, the concentration of uranium-235 decreases. This reduces the number of fissions occurring with a given number of neutrons. Furthermore, some of the fission products produced absorb neutrons, thus reducing the number of neutrons available to produce fission. These changes can be accommodated by withdrawing the control rods from the reactor and allowing more neutrons to be available in the fuel. After a long period of operation, however, such changes can no longer be accommodated, and the fuel may have become depleted in U-235 to the point where a continuous chain reaction can no longer be sustained. At this point, the reactor has to be refuelled with fresh fuel.

With reactors that are partially refuelled once a year, the control rods do not provide an adequate range of control, and therefore a soluble neutron absorber is added in small quantities to the moderator. Its concentration is gradually reduced over time to compensate for fuel burnup. Some reactors are designed for continuous on-load refuelling. This is advantageous because the effects of fuel burnup and fission product production are negligible with regard to overall reactor conditions.

5 Safety and Licensing

5.1 Radiation Hazards

Many types of radiation, both natural and human generated, are encountered in everyday life. Even the human body contains radioactive materials. This radiation can be divided into two categories. The first is electromagnetic radiation, which is assumed to travel in waves of discrete frequencies, for example, X-rays, γ-rays, and UV rays. The second is particulate radiation consisting of high-energy particles moving at high velocity, such as α-particles, β-particles, neutrons, etc. Charged particles such as α-particles and β-particles are easily stopped by ionization of surrounding material, but γ-rays and neutrons can be very penetrating due to their low interaction with most materials.

Radiation causes damage to biological structures such as cells. However, cells are in a continual state of dying and being replaced, so such damage is naturally repaired provided that it is not excessive. Of concern though is possible damage to DNA structures and subsequent genetic effects. For this reason exposure to radiation is controlled much more rigidly than exposure to toxic materials and dangerous chemicals. Although γ-rays and neutrons can penetrate the human body, α-radiation and β-radiation are generally damaging only if materials that emit these particles are ingested or come into contact with skin.

In the core of a nuclear reactor, the fission process produces neutrons and highly reactive fission products which emit α-particles and β-particles as well as γ-radiation as they decay to stable isotopes. Neutrons and γ-rays are emitted from the core, but are attenuated to low and safe levels by adequate shielding around the reactor. The fission products are held within the fuel and therefore do not pose a hazard unless released to the environment through leakage from the fuel or rupture of the fuel.

During operation of a nuclear plant, small amounts of leakage inevitably occur because some fission products are gaseous and attenuation cannot eliminate all radiation. Therefore, the entire operational area within and surrounding the plant as well as the local environment is continuously monitored to ensure that radiation is kept well within safe limits and usually kept much lower.

In the event of a serious accident, for example, fuel overheating and rupture of reactor components, the design of all reactors makes provision for barriers to contain these fission products and prevent their release to the environment. Such barriers may not be perfect, and some radioactive products may be released. If people in the surrounding area ingest such products, they will be subject to α-radiation or β-radiation emitted by these products, along with some γ-radiation resulting from the decay process. The design of the plant has, as a key objective, the provision of safety devices and containment barriers to ensure that radiation exposure to the public is kept well below acceptable limits in the event of an accident.

5.2 Risk Assessment

Humankind is subject to many risks, both natural and human generated. These risks have been analyzed, and if the severity of an accident is compared with the probability of its occurrence, it is generally found that the probability of an accident occurring is much higher if the number of deaths per accident is low (motor vehicle accidents) than if the number of deaths is high (aircraft accidents). Because the consequences of a serious nuclear accident (for example, the Chernobyl accident) can be very severe, the probability of its occurrence must be kept very low. This puts great pressure on the nuclear industry to minimize the risk of serious accidents to levels far below those of other industries. The result is that the nuclear industry is the most regulated of all industries. Highly sophisticated design and accident analysis techniques and requirements are in place to ensure public safety.

5.3 Licensing Principles

All nuclear plants are subject to a rigorous licensing process by an independent regulatory authority to ensure safety of operating personnel and the general public and to ensure that there are no detrimental effects to the surrounding environment. All aspects of the plant are checked and assessed with regard to the effects of possible failures of key components. Initially this assessment was based on the maximum credible accident due to a single failure. In the event of the worst possible accident, the likely damage to the plant was assessed and the containment barriers reviewed to determine the likely amount of radioactive fission product release to the environment. This release was to be kept below specified values. This concept did not take account of the possible (though very unlikely) failure of a component deemed not to fail, nor did it take account of multiple minor failures which could have a cumulative or cascading effect, causing an otherwise unpredicted failure of a key component.

The modern approach, based on the probability of failure of a key component and the consequences of such a failure, overcomes the shortcomings mentioned above. A component deemed not to fail can be given a very low but still finite probability of failure. Moreover, multiple components which are all associated with one type of accident can each be given a certain probability of failure, and hence the probability of simultaneous failure can be assessed. Ultimately the entire plant can be assessed to determine the probability of an accident leading to a certain release of radioactive fission products. Hence, the safety of the surrounding environment and population can be ensured to a very high degree, that is, a very low probability of damage to the environment or excessive radiation exposure to the population.

The transition to a probabilistic approach as opposed to the deterministic method was initially difficult due to a lack of statistical data and because certain failures could have different root causes. However, as the industry has matured, more statistical data have become available from test results and operating experience. Nevertheless, the process remains complex.

Another consideration is the relative importance of component failure and human error. A quick analysis done many years ago indicated that roughly half of various incidents and accidents in operating nuclear plants around the world were caused by human error and roughly half by component failure. Many component failures could be ascribed to human error during component design or manufacture. Rigorous quality control, quality assurance, and quality management procedures can minimize manufacturing and construction defects. Proper training, assessment, and licensing of plant operators can minimize human error during operation. This all contributes to a lower risk of accident and increased safety in nuclear plants.

A further development in the nuclear industry is the concept of passive safety. This means that in the event of a failure leading to accident conditions, the reactor will naturally revert to a safe shutdown condition and maintain that condition with minimal operator intervention, even in the event of the loss of key services. Some newer designs, for example, can maintain reactor cooling after shutdown by natural coolant circulation and natural heat convection to the environment.

6 Problems

1 Explain why the fission of very heavy elements results in release of neutrons as well as production of excess energy.

2 Explain the purpose of the key components of a nuclear reactor and show how these are arranged to ensure adequate heat removal and satisfactory control.

3 Describe the key similarities and differences between a CANDU and a PWR as well as between a CANDU and a BWR.

4 Explain how fission heat is converted into electrical energy and identify all key interfaces and points of energy conversion in the process.

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